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PROJECT NUMBER: 690

Dear Dr. Kuo:

We have attached NEI 95-10, Revision 6 *Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule* for your review. We appreciate the opportunities to discuss with your staff the issues related to this revision and have appropriately addressed those issues. We believe this revision of 95-10 resolves the exceptions taken in Draft Regulatory Guide DG-1140 and look forward to your endorsement.

Please contact me (202-739-8080; am@nei.org) or James Ross (202-739-8101; jr@nei.org) with any questions regarding this transmittal.

Sincerely,

Alex Marion

Alex Marion

Attachment

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D042

Rec'd 7/5/05

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Industry Guideline For Implementing The Requirements of 10 CFR Part 54 – The License Renewal Rule

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ACKNOWLEDGEMENTS

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NEI also wishes to express its appreciation to the Electric Power Research Institute that devoted considerable time and resources to the development of this industry guideline.

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ACRONYMS

AMP	Aging Management Program
AMR	Aging Management Review
ARDM	Age Related Degradation Mechanism
BTP	Branch Technical Position
CLB	Current Licensing Basis
CRGR	Committee to Review Generic Requirements
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
FAC	Flow-Accelerated Corrosion
FP	Fire Protection
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GSI	Generic Safety Issue
HELB	High Energy Line Break
I&C	Instrumentation and Controls
IEEE	Institute of Electronic and Electrical Engineers
IPA	Integrate Plant Assessment
IPEEE	Individual Plant Examination of External Events
LRA	License Renewal Application
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OMB	Office of Management and Budget
PD-RLEP	Program Director, License Renewal & Environmental Impacts Program
PRA	Probabilistic Risk Assessment
RAI	Request for Additional Information
SAMG	Severe Accident Management Guidelines
SER	Safety Evaluation Report
SOC	Statement of Considerations
SQUG	Seismic Qualification Utility Group
SR	Safety Related
SRP-LR	Standard Review Plan – License Renewal
SSC	Systems, Structures, and Components
TLAA	Time-Limited Aging Analysis
UFSAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue

GUIDELINE FOR IMPLEMENTING THE REQUIREMENTS OF 10 CFR PART 54 -THE LICENSE RENEWAL RULE

1 INTRODUCTION

This guideline provides an acceptable approach for implementing the requirements of 10 CFR Part 54, the license renewal rule, hereinafter referred to as the Rule. The process outlined in this guideline is founded on industry experience in implementing the license renewal rule. It is expected that following this guideline will offer a stable and efficient process, resulting in the issuance of a renewed license. However, applicants may elect to use other suitable methods or approaches for satisfying the Rule's requirements and completing a license renewal application.

This guideline uses terminology specific to the license renewal rule. A copy of 10 CFR Part 54 is provided as Appendix A and should be reviewed.

1.1 Background

In December 1991, the Nuclear Regulatory Commission (NRC) published 10 CFR Part 54 to establish the procedures, criteria and standards governing nuclear plant license renewal. Since publishing the original rule, the NRC and the industry conducted various activities related to its implementation. In September 1994, the NRC proposed an amendment to the Rule. The final amendment was published in May 1995. It focuses on the effects of aging on long-lived passive structures and components and time-limited aging analyses (TLAAs) as defined in 10 CFR 54.21(a)(1) and 54.3, respectively. In addition, the amendment allows greater reliance on the current licensing basis (CLB), the maintenance rule and existing plant programs.

1.2 Purpose and Scope

The major elements of the guideline (with their respective guideline sections) include:

- Identifying the systems, structures and components within the scope of license renewal (Section 3.1);
- Identifying the intended functions of systems, structures and components within the scope of license renewal (Section 3.2);
- Identifying the structures and components subject to aging management review and intended functions (Section 4.1);
- Assuring that effects of aging are managed (Section 4.2);
- Application of new programs and inspections for license renewal (Section 4.3);

- Identifying and resolving time-limited aging analyses (Section 5.1);
- Identifying and evaluating exemptions containing time-limited aging analyses (Section 5.2); and
- Identifying a standard format and content of a license renewal application (Section 6.0).

Applicants interested in license renewal are responsible for preparing a plant-specific license renewal application. The license renewal application includes general information and technical information. The general information is much the same as that provided with the initial operating license application. The technical information includes an Integrated Plant Assessment, the CLB changes during the NRC review of the application, TLAAs, a supplement to the Final Safety Analysis Report, any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation and a supplement to the plant's environmental report that complies with the requirements of Subpart A of 10 CFR Part 51.

1.3 Applicability

This document is applicable to any operating license for nuclear power plants licensed pursuant to Sections 103 or 104b of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

1.4 Utilization of NUREG-1800, NUREG-1801, Regulatory Guide 1.188 and NRC Interim Staff Guidance Documents

Applicants should consider three regulatory guidance documents: the Generic Aging Lessons Learned Report, NUREG-1801, the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800 and Regulatory Guide 1.188, Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses. NUREG-1801 documents the NRC staff evaluation of generic aging management programs to document the basis for determining when such programs are adequate without change and when they should be augmented for license renewal. NUREG-1801 is a basis document to NUREG-1800 that provides NRC staff guidance in reviewing a license renewal application. Regulatory Guide 1.188 provides guidance on the format for the information that is to be submitted in a license renewal application.

NUREG-1801 contains tables with various nuclear power plant components, materials, environments, aging effects/mechanisms, aging management programs and a column noting whether evaluation beyond that contained in NUREG-1801 is required. It also contains the NRC evaluation of common aging management programs. Many of these programs have been determined adequate, without change, to manage aging effects for particular structures and components. These include programs commonly credited for managing aging effects associated with time-limited aging analyses. There are evaluations of mechanical programs, structural programs and electrical programs. NUREG-1801 programs are one acceptable way to manage aging effects. An applicant may credit other programs for license renewal. NUREG-1801 has an appendix that discusses quality assurance for aging management programs.

NUREG-1800 contains guidance for NRC reviewers of license renewal applications. Its principal purpose is to ensure the quality and uniformity of staff reviews of applications. It contains a chapter corresponding to each of the sections of an application: administrative information, scoping and screening methodology for identifying structures and components subject to aging management review and implementation results, aging management review results and time-limited aging analyses. An appendix contains three branch technical positions (BTP). BTP RLSB-1 addresses the aging management demonstration required by 10CFR54.21(a)(3). BTP IQMB-1 describes an acceptable process for implementing the corrective actions, the confirmation process and the administrative controls elements of aging management programs for license renewal. BTP RLSB-2 addresses aging effects or time-limited aging analyses related to unresolved safety issues or generic safety issues.

Regulatory Guide 1.188 provides a summary of application contents and formatting specifications.

Changes and clarifications to the above guidance documents suggested by license renewal stakeholders and approved by the staff can be communicated via interim staff guidance (ISG) documents. The process is described in a December 21, 2001, NRC letter, ISG-8. ISGs that have not been incorporated into license renewal guidance documents should be considered by applicants. Details about each of the ISGs are available on the NRC License Renewal Guidance Documents web page.

Generally ISGs will discuss technical issues rather than process issues.

It is important for applicants to note that the ISG positions may require, for license renewal, considerations that differ from the applicant plant's current licensing basis. Applicants may want to ensure their applications are clear with respect to the current licensing basis and note some application content is based on an ISG rather than the CLB.

1.5 Resolution of Current Safety Issues (e.g., GSIs and USIs)

Generic resolution of a generic safety issue (GSI) or unresolved safety issue (USI) is not necessary for the issuance of a renewed license. GSIs and USIs that do not contain issues related to the license renewal aging management review or time-limited aging evaluation need not be reviewed. However, designation of an issue as a GSI or USI does not exclude the issue from the scope of the aging management review or time-limited aging evaluation.

Unresolved Safety Issues, HIGH and MEDIUM priority issues described in Appendix B in NUREG-0933, that involve aging effects for structures and components subject to an aging management review or TLAAs, should be specifically addressed. The version of NUREG-0933 that is current on the date six months before the date of the license renewal application should be used to identify such issues. Prior to Safety Evaluation Report completion, any new issues contained in later versions of NUREG-0933 must be reviewed and addressed if determined to be applicable to the applicant's plant and they involve aging effects for structures and components subject to aging management review or they are associated with a time-limited aging analysis. The results may be submitted to NRC in the annual update.

For a GSI or USI affecting the aging management review or time-limited aging evaluation, there are several approaches that can be used to satisfy the finding required by §54.29.

- If resolution has been achieved before issuance of a renewed license, implementation of that resolution could be incorporated within the renewal application. The plant-specific implementation information should be provided.
- An applicant may choose to submit a technical rationale that demonstrates that the CLB will be maintained until some later time in the period of extended operation, at which time one or more reasonable options (e.g., replacement, analytical evaluation or a surveillance/maintenance program) would be available to adequately manage the effects of aging. The license renewal application would have to describe the basis for concluding that the CLB is maintained in the period of extended operation and briefly describe options that are technically feasible during the period of extended operation to manage the effects of aging, but it would not have to pre-select which option would be used.
- Another approach could be for an applicant to develop an aging management program that, for that plant, incorporates a resolution to the aging effects issue.
- Another option could be to propose to amend the CLB (as a separate action outside the license renewal application), which, if approved, would remove the intended function(s) from the CLB.

During the preparation and review of a renewal application, an applicant or the NRC may become aware of an aging management or time-limited aging analysis issue that may be generically applicable (but is not yet part of the formal generic safety issue resolution process). An applicant must still address the issue in its application to demonstrate that the effects of aging are or will be adequately managed or that TLAAs have been evaluated for the period of extended operation.

See NUREG-1800 Appendix A.3, BTP RLSB-2 for more information on this matter.

1.6 Organization of the Guideline

Obtaining a renewed operating license is a three-phase approach. The first phase is the technical work that must be performed to generate the information that is included in the license renewal application. The second phase is the preparation of the license renewal application. Phase three is submitting the application and the post-submittal activities.

The technical work includes determining the systems, structures and components within the scope of the Rule, identifying the structures and components subject to an aging management review, identifying aging effects requiring management, evaluating plant programs, and reviewing TLAAs and exemptions and justifying their applicability for license renewal. The technical phase produces results or information that is ultimately incorporated into the license renewal application, so it is important to maintain accurate and detailed supporting documentation. This supporting documentation is not required to be submitted as part of the application; however, it must be auditable and retrievable for NRC review. Sections 3, 4 and 5 of this document provide guidance on how to proceed through the technical phase. These sections explain what work needs to be done, how to do it and the expected results.

Section 6 discusses the standard license renewal application format. The standard format is shown in Appendix D.

Section 7 discusses the activities after submittal of the application including annual updates, the review and post-renewal process requirements.

Earlier versions of NEI 95-10 included examples to illustrate the different steps involved in preparing a license renewal application. The examples are no longer included. Instead, applicants are encouraged to review applications that have been submitted and the resulting safety evaluation reports that are issued in the form of NUREGs.

2 OVERVIEW OF PART 54

The Rule contains the regulatory requirements that must be satisfied in order to obtain a renewed operating license, which allows continued operation of a nuclear power plant beyond its original license term. (Figure 2.0-1 reflects the license renewal implementation process.)

The Rule is founded on two principles. The first principle of license renewal is that with the possible exception of the detrimental effects of aging on the functionality of certain plant systems, structures and components in the period of extended operation and possibly a few other issues related to safety only during the period of extended operation, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. The second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term.

In addition to the identification and evaluation of time-limited aging analysis (TLAAs), the focus of the Rule is on providing reasonable assurance that the effects of aging on the functionality of long-lived passive structures and components are adequately managed in accordance with the plant- specific current licensing basis (CLB) design basis conditions such that the intended functions are maintained in the period of extended operation. This demonstration is documented in the license renewal application.

The license renewal application contains general information, technical information, information regarding technical specifications and environmental information.

The general information concerns the plant site and the plant owner(s). The required information is specified in 10 CFR 50.33(a) through (e), (h) and (i). Additionally, the application must include conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license.

The technical information includes (1) the integrated plant assessment (IPA), which is the demonstration that the effects of aging on long-lived, passive structures and components are being adequately managed such that the intended functions are maintained, consistent with the CLB, in the renewal period, (2) the listing and evaluation of TLAAs and any exemptions in effect that are based on TLAAs and (3) a supplement to the plant's FSAR that contains a summary description of the programs and activities that are cited as managing the effects of aging and the evaluation of time-limited aging analyses.

The application also must include any changes or additions to the plant's technical specifications that are necessary to manage the effects of aging during the period of extended operation. Last, the application must contain a supplement to the plant's environmental report that complies with the requirements of 10 CFR Part 51.

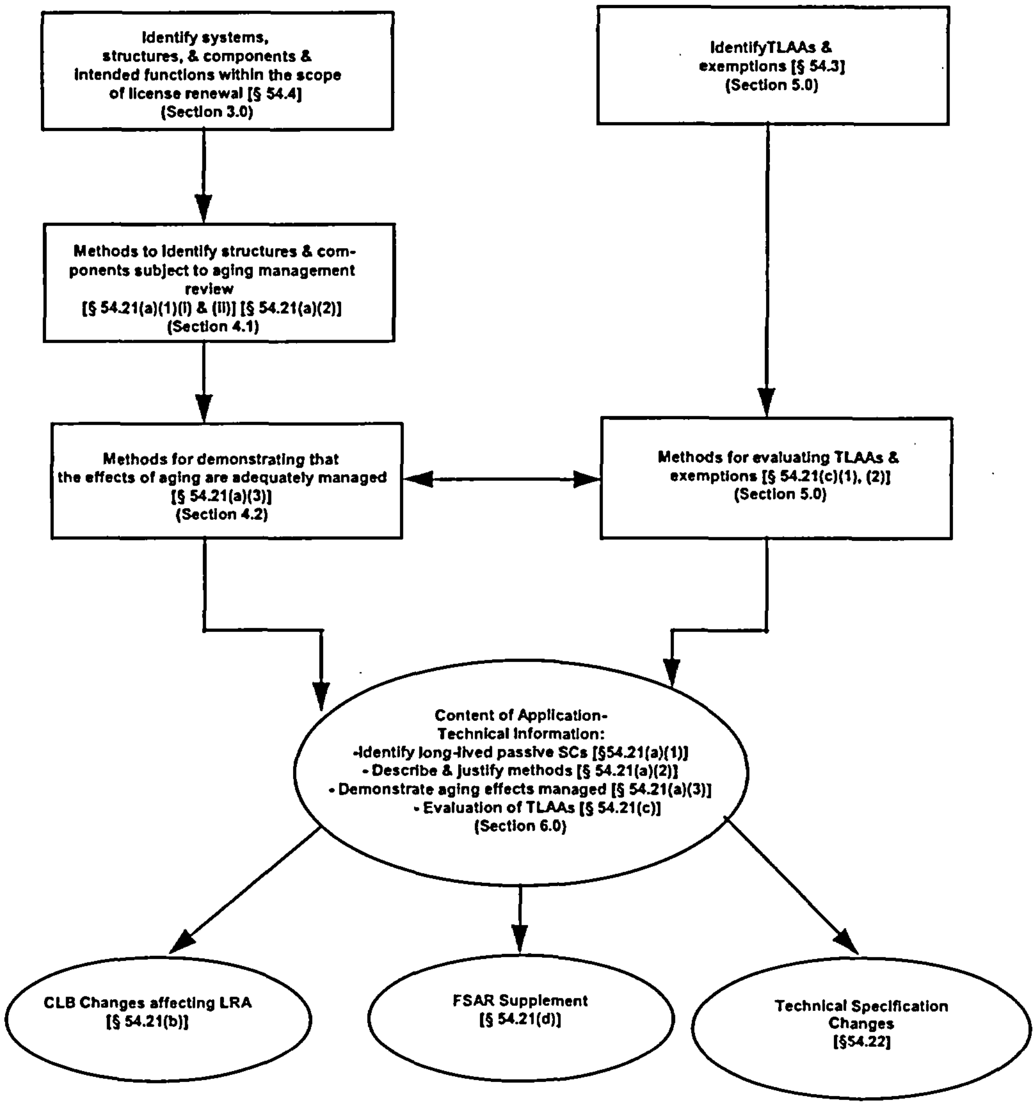
Once the application is submitted to the NRC, it must be amended each year to identify any changes to the CLB that materially affect the contents of the application, including the FSAR supplement.

Information and documentation required by, or otherwise necessary to document compliance with, the Rule must be maintained by the applicant in an auditable and retrievable form for the term of the renewed operating license. Additionally, after the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures or components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with §54.21.

The license renewal rule at 10 CFR 54.30 specifies matters that are not subject to NRC review and that may not be contested in a hearing for license renewal. The intent of the provision in 10 CFR 54.30 is to clarify that safety matters of noncompliance for the current operating term should not be the subject of the renewal application or the subject of a hearing in a renewal proceeding, absent specific NRC direction. Issues concerning operation during the currently authorized term of operation should be addressed as part of the current license in accordance with the Commission's current regulatory process rather than deferred until a renewal review (which will not occur if the licensee chooses not to renew its operating license). Furthermore, 10 CFR 54.30 is intended to make clear that aging issues discovered during the renewal review for the structures and components that are reviewed in 10 CFR 54.21(a)(3) or 54.21 (c)(1) and that raise questions about the capability of these structures and components to perform their intended function during the current term of operation must be addressed under the current license. However, an applicant for renewal is not relieved from addressing the issue relevant to the period of extended operation as part of its renewal application.

Section 54.30 does not require a general demonstration of compliance with the CLB as a prerequisite for issuing a renewed license. Section 54.30 discusses the applicant's responsibilities for addressing safety matters under its current license, which are not within the scope of the renewal review.

FIGURE 2.0 -1
LICENSE RENEWAL IMPLEMENTATION PROCESS



3 IDENTIFY THE SSCs WITHIN THE SCOPE OF LICENSE RENEWAL AND THEIR INTENDED FUNCTIONS

This section provides a process for determining which of the many systems, structures and components that make up a commercial nuclear power plant are included within the scope of the Rule. The scoping process described in this guideline is at the system and structure level for the majority of the systems, structures and components. This is not intended to imply that scoping at a component level is not allowed by the Rule. In fact, for some plants it may be easier to scope at the component level. In addition, it may be convenient for a plant to scope using more than one method. For instance, a system-based scoping approach may be used for mechanical systems and a component or commodity-based scoping approach used for electrical systems. (Figure 3.0-1 is a process diagram for this section.)

To assist the applicant in determining the systems, structures and components within the scope of license renewal a list of potential information sources is provided as Table 3.1.1. The table is not intended to be all encompassing nor is it intended to be a list of "must review" sources. During the development of this guidance document, there was significant interaction with the NRC staff regarding the inclusion of probabilistic risk assessment (PRA) summary report and individual plant examination of external events (IPEEE) in the table. Clearly, these two sources contain information that is beyond the plants' licensing basis, and if the applicant chooses to use them as information sources, ultimately, the provisions of § 54.4 prevail. This means that while the PRA summary report and the facility's IPEEE may mention systems structures and components, only those that meet the criteria delineated in § 54.4 are considered in the license renewal scope.

The Commission was clear on this point in the Statements of Consideration for the 1995 license renewal rulemaking. In response to a comment from the state of Illinois, the Commission acknowledges the existence of the PRA and IPEEEs; however, the Commission also stated "The CLB for currently operating plants is largely based on deterministic engineering criteria. Consequently, there is considerable logic in establishing license renewal scoping criteria that recognize the deterministic nature of a plant's licensing basis. Without the necessary requirements and appropriate controls for plant-specific PRAs, the Commission concludes that it is inappropriate to establish a license renewal scoping criterion, as suggested by Illinois, that relies on plant-specific probabilistic analyses."

The table also identifies the emergency operating procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) as potential information sources. Like the PRA summary report and the IPEEE studies, the EOPs and SAMGs are beyond design basis. While the Commission did not speak to the use of these documents in the Statements of Consideration, it is reasonable to extend the Commissions view on the use of PRA and IPEEEs as scoping criteria to the EOPs and SAMGs as well.

NUREG-1800 section 2.1.3 and Table 2.1-1 list the documents an NRC reviewer is expected to consider. NUREG-1800 section 2.1.3.1 describes some general expectations of scoping. NUREG-1800 sections 2.2.1, 2.2.3.1 and Table 2.2-1 provide examples an NRC reviewer may consider in reviewing scoping results.

3.1 Systems, Structures and Components Within the Scope of License Renewal

Part 54 Reference

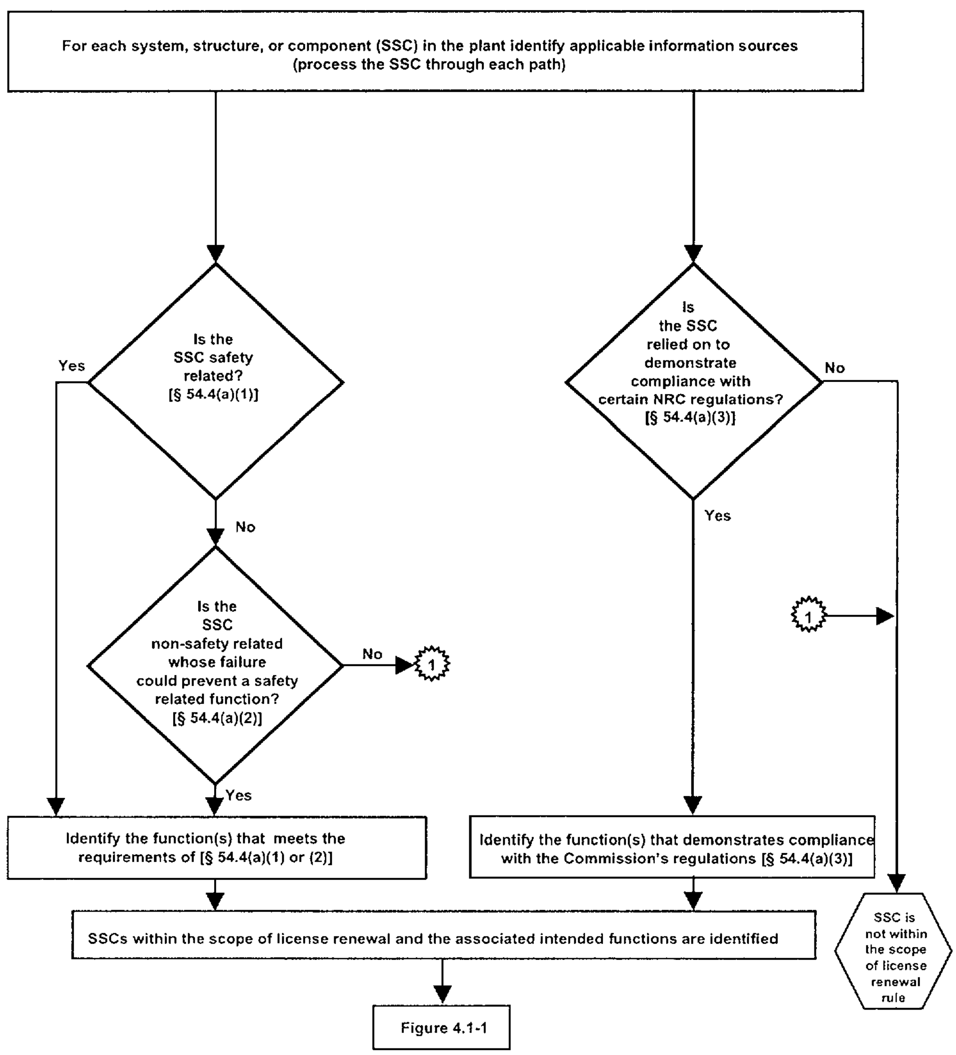
§54.4
(a) Plant systems, structures, and components within the scope of this part are -

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined as in 10 CFR 50.49 (b)(1)) to ensure the following functions -
(i) The integrity of the reactor coolant pressure boundary;
(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1), 50.67(b)(2), or § 100.11 of this chapter, as applicable.

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

FIGURE 3.0-1
A METHOD TO IDENTIFY SSCs AND INTENDED FUNCTIONS WITHIN THE SCOPE OF LICENSE RENEWAL [§ 54.4(a) &(b)]



3.1.1 Safety-Related Systems, Structures and Components

There are a number of viable alternatives for identifying safety-related systems, structures and components. Table 3.1-1 is a listing of information sources for consideration in this process. There may be information sources available to applicants that are not identified on Table 3.1-1. These sources may be considered as well.

Regardless of the approach used, a safety-related system, structure or component is within the scope of license renewal if it is relied upon to remain functional during and following design basis events as defined in §50.49(b)(1) to ensure the following functions:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposure comparable to the guidelines in § 50.34(a)(1) § 50.67(b)(2) or § 100.11 of this chapter, as applicable.

It is conceivable that, because of plant unique considerations and preferences, applicants may have previously elected to designate some systems, structures and components as safety related that do not perform any of the requirements of §54.4(a)(1). Therefore, a system, structure or component may not meet the requirements of §54.4(a)(1) although it is designated as safety related for plant-specific reasons. However, the systems, structures and components would still need to be evaluated for inclusion into the scope of the Rule using the criteria in §54.4(a)(2) and §54.4(a)(3). For example, an applicant may have designated refueling equipment as safety related even though it does not meet the criteria delineated above. In such cases, the applicant shall include a discussion of the process (in accordance with §54.21(a)(2)) for making these determinations.

Similarly, an applicant's current licensing basis (CLB) definition of safety related may not match the §54.4(a)(1) definition. In these cases, the applicant should apply the §54.4(a)(1) definition for purposes of identifying the systems, structures and components that are in the scope of license renewal. This is consistent with NUREG-1800 section 2.1.3.1.1.

3.1.2 Nonsafety-Related SSCs Whose Failure Prevents Safety-Related SSCs From Fulfilling Their Safety-Related Function

An applicant should rely on the plant's CLB, actual plant-specific experience, industry wide operating experience, as appropriate, and existing plant-specific engineering evaluations to determine the appropriate systems, structures and components in this category. Consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required. Hypothetical failures that are part of the CLB may require consideration of second-, third- or fourth-level support systems. NUREG-1800 section 2.1.3.1.2 and Table 2.1-2 contain NRC expectations on nonsafety-related scoping. See Appendix F for the industry guidance for §54.4(a)(2) scoping criterion. Also see NUREG-1800 Table 2.1-2 regarding hypothetical failures.

3.1.3 SSCs Relied on to Demonstrate Compliance With Certain Specific Commission Regulations

Systems, structures and components relied on to perform a function that demonstrates compliance with the following regulations are also in the scope of the Rule:

- Fire Protection (10 CFR 50.48)
- Environmental Qualification (10 CFR 50.49)[1]
- Pressurized Thermal Shock (10 CFR 50.61)
- Anticipated Transient Without Scram (10 CFR 50.62)
- Station Blackout (10 CFR 50.63)

The information sources in Table 3.1-1 could be considered for identifying the systems, structures and components whose functions are relied on to demonstrate compliance with the regulatory requirements (i.e., whose functions were credited in the analysis or evaluation). Mere mention of a system, structure or component in the analysis or evaluation does not constitute support of a specified regulatory function. An applicant should rely on the plant's CLB, plant-specific experience, industry wide operating experience, as appropriate and existing plant-specific engineering evaluations to determine the appropriate systems, structures and components in this category. Consideration of hypothetical failures that could result from system interdependencies that are not part of the plant's CLB and that have not been previously experienced is

[1] The Statements of Consideration for the amendments to 10 CFR Part 54[60FR22466] states that "...the Commission agrees that for purposes of §54.4, the scope of §50.49 equipment to be included within §54.4 is that equipment already identified by licensees under 10 CFR 50.49(b). Licensees may rely upon their listing of 10 CFR 50.49 equipment, as required by 10 CFR Part 50.49(d), for purposes of satisfying §54.4 with respect to equipment within the scope of §50.49."

not required. Hypothetical failures that are part of the CLB may require consideration of second-, third- or fourth-level support systems. See NUREG-1800 section 2.1.3.1.3 for NRC expectations on regulated events scoping. Also see Table 2.1-2 regarding cascading.

TABLE 3.1-1

SAMPLE LISTING OF POTENTIAL INFORMATION SOURCES

• Verified Databases (database that is subject to administrative controls to assure and maintain the integrity of the stored data or information)
• Master Equipment Lists (including NSSS Vendor Listings)
• Q-Lists
• Updated Safety Analysis Reports
• Piping and Instrument Diagrams
• Electrical One-Line or Schematic Drawings
• Operations and Training Handbooks
• Design Basis Documents
• General Arrangement or Structural Outline Drawings
• Quality Assurance Plan or Program
• Maintenance Rule Compliance Documentation
• Design Basis Event Evaluations
• Docketed Correspondence
• System Interaction Commitments
• Technical Specifications
• Environmental Qualification Program Documents
• Regulatory Compliance Reports (Including Safety Evaluation Reports)
• Probabilistic Risk Assessment Summary Report
• Emergency Operating Procedures
• Severe Accident Management Guidelines
• Individual Plant Examination of External Events

3.2 Intended Functions of SSCs Within the Scope of License Renewal

Part 54 Reference

§54.4

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(b) The intended functions that these systems, structures, and components must be shown to fulfill in §54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1)-(3) of this section.

The intended functions define the plant process, condition or action that must be accomplished in order to perform or support[2] a safety function for responding to a design basis event or to perform or support a specific requirement of one of the five regulated events in §54.4(a)(3). At a system level, the intended functions may be thought of as the functions of the system that are the bases for including this system within the scope of license renewal as specified in §54.4(a)(1)-(3). Where the plant's licensing basis includes requirements for redundancy, diversity and defense-in-depth, the system intended functions include providing for the same redundancy, diversity and defense-in-depth during the period of extended operation. For example, a system with two independent trains, according to the plant's CLB, has to perform the intended functions by each independent train.

As noted in the above reference, §54.4(b) provides criteria that should be used to identify the intended functions of systems, structures and components within the scope of the Rule. Therefore, as part of the license renewal process, an applicant should establish a method that identifies systems, structures and components within the scope of the Rule and the intended functions that are the basis for their inclusion.

In identifying intended functions it is important to understand that the terms "systems, structures and components" and "structures and components" are used differently throughout the Rule and statement of consideration (SOC). The SOC, in a footnote (60FR22462), clarifies why "systems, structures and components" is used in some sections of the SOC and Rule versus "structures and components ". The SOC clarifies that the scoping section (§54.4) includes systems, structures and components rather than just structures and components to allow an applicant flexibility in how it develops and implements a methodology to identify those structures and components that are subject to an aging management review for license renewal. Also, §54.4 and the associated SOC sections include systems,

[2]The term "support" here includes system, structure and components whose failure could prevent other SSCs from performing their intended function.

structures and components to allow the applicant flexibility on how exemptions containing TLAAs can be evaluated for the period of extended operation (§54.21 (c)(2)) because exemptions might have been granted for a particular system.

The integrated plant assessment (IPA) required by §54.21(a) is performed at the structure and component level. Guidance on the IPA process is provided in Section 4 of this guideline. The Rule contains flexibility to permit an applicant to start the IPA process at either the system/structure or structure/component level as long as the passive, long-lived structures and components are identified. The intended functions of the structures and components are the same regardless of the starting point. If the starting point is the system level, the system intended functions are identified as previously discussed. However, the intended functions of the structures and components still have to be determined as discussed in Section 4.1. These functions are the specific functions of the structures and components that support the system/structure intended function(s). Similarly, if the starting point is the structure and component level, the intended functions are those that included these structures and components within the scope of license renewal. A structure or component may have multiple functions, but only the function(s) meeting the criteria of §54.4 are to be identified for license renewal. See NUREG-1800 Table 2.1-3 for an example. Intended functions need not be defined for component piece-parts.

The process leading to the maintenance rule scoping determinations may also have produced a listing of the system and structure functions. Although it is not a requirement of the maintenance rule, such a listing may be based on a documented procedure that ensures a comprehensive and consistent approach to defining the functions for all the systems within the scope of the maintenance rule. If this is the case, then the maintenance rule documentation can be used to help identify the functions of safety-related systems and nonsafety-related (affecting safety-related) systems within the scope of the license renewal rule. The information sources used to identify the systems required for compliance with the regulations in §54.4(a)(3) should be used to identify their associated functions. If the maintenance rule documentation does not define the system functions, does not rely on a procedure that uses a structured approach or the applicant elects not to use this source, then alternative documentation such as a verified database or a safety analysis report, operations training manuals, etc., can be used to identify the functions of safety-related systems and nonsafety-related (affecting safety-related) systems. A sample listing of information sources that can be used to identify the functions of all systems (and structures and components) within the scope of the Rule is provided in Table 3.1-1.

3.3 Documenting the Scoping Process

Section 54.37(a) of the Rule requires applicants to retain in an auditable and retrievable form all information and documentation required by, or otherwise necessary to document compliance with, the provisions of the Rule.

The results of the scoping determination should be documented in a format consistent with other plant documentation practices. The information may be maintained in hard-copy or electronic format. If available and appropriate, the information may be incorporated into an existing plant database. The applicant should use the quality assurance program in effect at the plant when documenting the results of the scoping process.

The information to be documented by the applicant should include:

A designation of the plant systems, structures and components that are safety related (§54.4 (a)(1)), meet the requirements of §54.4(a)(2), or meet the requirements of §54.4(a)(3);

Identification of the systems', structures' and components' functions that meet the requirements of §54.4(b) and therefore are intended functions; and

The information sources, used to accomplish the above, and any discussion needed to clarify their use.

NRC inspection for compliance with this requirement is performed in accordance with Inspection Procedure 71002, License Renewal Inspection.

Applicants have typically provided mechanical system drawings to NRC concurrent with the application. The drawings are generally not a part of the application and are submitted only to facilitate NRC staff review. The NRC staff reviews the scoping and screening in accordance with Section 2.3 of the Standard Review Plan for License Renewal (NUREG-1800). To facilitate NRC staff review, the applicants should submit drawings showing the mechanical components that are within the scope of license renewal in accordance with 10CFR 54.4(a), and in addition, system functions that meet 10CFR 54.4(a) should also be identified.

4 INTEGRATED PLANT ASSESSMENT

The integrated plant assessment (IPA) is the core of the license renewal application. It is the transition from the scoping process to the screening process where the focus is on components and structures and their intended functions. Once the systems, structures and components within the scope of license renewal are identified, the next step is to determine which structures and components are subject to an aging management review. Specifically, §54.21(a)(1) states that the aging management review is required for the structures and components that perform an intended function without moving parts or without a change in configuration or properties (i.e., it is passive) and that are not subject to replacement based on a qualified life or specified time period (i.e, it is long-lived). The IPA also includes a description and justification of the methods used to determine the passive, long-lived structures and components and a demonstration that the effects of aging on those structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the plant-specific CLB for the period of extended operation.

Section 4.1 presents one method to identify structures and components subject to aging management review. There are two steps required to perform an aging management review. First, aging effects that require management are identified and evaluated. Then aging management programs are identified to manage the effects of aging such that the intended component or structure function can be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Section 4.2 describes methods to identify aging effects requiring management. Evaluation of aging management programs is presented in Section 4.3.

4.1 Identification of Structures and Components Subject to Aging Management Review and Intended Functions

Part 54 Reference

§54.21(a)(1)(i) and (ii)

(1) For those systems, structures, and components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components -

(i) That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

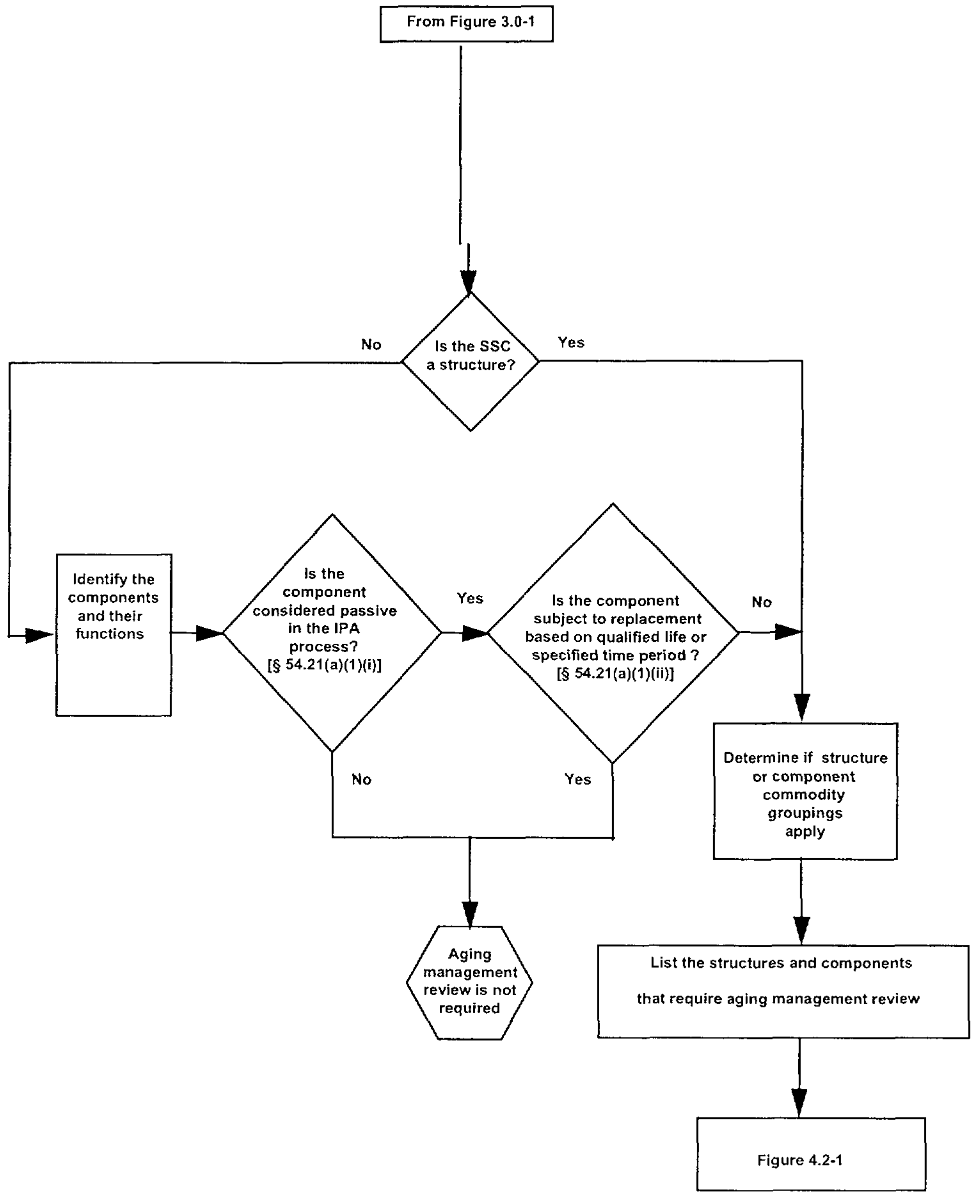
(ii) That are not subject to replacement based on a qualified life or specified time period.

§54.21(a)(2)

(2) Describe and justify the methods used in paragraph (a)(1) of this section.

The method that will accomplish the objective of identifying structures and components subject to aging management review must identify the structures and components meeting the criteria of §54.21(a)(1)(i) and (ii). (Figure 4.1-1 reflects the method described in this section.)

FIGURE 4.1-1
IDENTIFICATION OF STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW [§ 54.21(a)(1)]



Selection of an appropriate method is dependent on the applicant's information management system(s). For example, the availability of computer databases of plant equipment may result in a more efficient component-by-component review. Absent such databases, an applicant may use a manual review based on system piping and instrumentation drawings and electrical one-line diagrams supplemented by other plant documentation as required.

If an applicant chooses, the applicant can use a bounding approach and the list could be larger (e.g., all passive structures and components). Such a bounding approach may be more efficient, especially when a program will cover all structures or components in an area whether or not all the structures or components in the area are subject to aging management review.

All long-lived passive structures and components that perform or support an intended function without moving parts or a change in configuration or properties are subject to aging management review. For all such structures or components, the structure or component intended function is documented for use during the aging management review of the IPA. The structure or component intended function is the specific function of the structure or component that supports the system intended function. Plant-specific CLBs require intended functions to be performed under a variety of design conditions. (Table 4.1-1 is a listing of typical passive structure and component intended functions.)

In making the determination that a structure's or component's intended function is performed without moving parts or a change in configuration or properties, it is not necessary to consider the piece parts of the structure or component. However, in the case of valves and pumps, the valve bodies and pump casings may perform an intended function by maintaining the pressure-retaining boundary and therefore would be subject to aging management review.

If the structure or component is not subject to replacement based on a qualified life or specified time period, then it is considered long-lived pursuant to §54.21(a)(1)(ii). Replacement programs may be based on vendor recommendations, plant experience or any means that establishes a specific service life, qualified life or replacement frequency under a controlled program. Structures and components that are <u>not long-lived</u> are not subject to aging management review.

<u>Use of Commodity Groups</u>
It may be beneficial to create commodity groups of like structures or components, possibly including those that are active and passive, to disposition the entire group with a single aging management review. The basis for group structures or components can be such characteristics as similar design, similar materials of construction, similar aging management practices and similar environments. If the environment in which the structure or component operates suggests potential different environmental stressors, then the commodity group determination also

could consider service time, operational transients, previous failures and any other conditions that would suggest different results. Appendix B of this guideline is a listing, although not all-inclusive, of typical plant components, structures and commodity groups, along with a determination of whether the group is active or passive. Applicants are encouraged to use this appendix in determining structures and components subject to an aging management review.

Structures Requiring Aging Management Review
Structures within the scope of license renewal are long-lived and passive and require aging management review. It may be useful, however, to categorize structures by type (e.g., poured concrete, block concrete, structural steel, shield walls, metal siding, foundation on piles, etc.) in preparation for the aging management review. Subdividing complex structures into discrete elements (e.g., walls, floors, slabs, doors, penetrations, foundations, etc.) may be useful because some elements may not have intended functions as defined in the Rule and, therefore, are not subject to aging management review. It may also be useful to individually identify spill containment, flood control and fire barrier structural components where applicable and appropriate. A building, for example, with several rooms may be in the scope of renewal because one of its rooms performs an intended function. Only that one room needs to be identified as requiring aging management review.

Structural Support Components
Structural supports either support or restrain mechanical and electrical equipment (e.g., hangers, pipe whip restraints, cable trays and supports). Structural supports can be considered part of or separate from the applicable structure. This guideline assumes that structural support commodity groups will be addressed separately from the applicable structure.

Also, there may be piping segments that provide structural support. For example, the safety-related/nonsafety-related boundary along a pipe run may occur at a valve location. The piping segment between this valve and the next seismic anchor provides structural support in a seismic event. This piping segment is within the scope of license renewal.

Complex Assemblies
Some structures and components, when combined, are considered a complex assembly (e.g., diesel generator starting air skids or heating, ventilating, and air conditioning refrigerant units). An applicant should establish the boundaries for such assemblies by identifying each structure and component that makes up the complex assembly and determining whether each one is subject to aging management review. NUREG-1800 Table 2.1-2 provides an example for a control room chiller assembly of how the components that require aging management review might be determined.

<u>Consumables</u>

Consumables also need to be considered in the process for determining the structures and components subject to an aging management review. Consumables, as used in this guideline, comprise the following four categories: (a) packing, gaskets, component seals, O-rings; (b) structural sealants; (c) oil, grease and component filters; (d) system filters, fire extinguishers, fire hoses and air packs. Table 4.1-2 and NUREG-1800 Table 2.1-3 provide methods to disposition these consumables. Disposition of consumables should be described in the methodology as noted in NUREG 1800, Table 2.1-3.

TABLE 4.1-1
TYPICAL PASSIVE STRUCTURE AND COMPONENT INTENDED FUNCTIONS

Intended Function	Description
Absorb Neutrons	Absorb neutrons
Electrical Continuity	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals
Insulate (electrical)	Insulate and support an electrical conductor
Filter	Provide filtration
Heat Transfer	Provide heat transfer (See Appendix C, Reference 1)
Leakage Boundary (Spatial)	Nonsafety-related component that maintains mechanical and structural integrity to prevent spatial interactions that could cause failure of safety-related SSCs
Pressure Boundary	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered, or provide fission product barrier for containment pressure boundary, or provide containment isolation for fission product retention
Spray	Convert fluid into spray
Structural Integrity (Attached)	Nonsafety-related component that maintains mechanical and structural integrity to provide structural support to attached safety-related piping and components
Structural Support	Provide structural and / or functional support to safety-related and/or nonsafety-related components
Throttle	Provide flow restriction

TABLE 4.1-1 (continued)
TYPICAL PASSIVE STRUCTURE AND COMPONENT INTENDED FUNCTIONS

Intended Function	Description
Direct Flow	Provide spray shield or curbs for directing flow (e.g., safety injection flow to containment sump)
Expansion/Separation	Provide for thermal expansion and/or seismic separation
Fire Barrier	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Flood Barrier	Provide flood protection barrier (internal and external flooding event)
Gaseous Release Path	Provide path for release of filtered and unfiltered gaseous discharge
Heat Sink	Provide heat sink during SBO or design basis accidents
HELB Shielding	Provide shielding against high energy line breaks
Missile Barrier	Provide missile barrier (internally or externally generated)
Pipe Whip Restraint	Provide pipe whip restraint
Pressure Relief	Provide over-pressure protection
Shelter, Protection	Provide shelter/protection to safety-related components
Shielding	Provide shielding against radiation
Shutdown Cooling Water	Provide source of cooling water for plant shutdown.
Structural Pressure Barrier	Provide pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events

TABLE 4.1-2
TREATMENT OF CONSUMABLES

Consumable	Disposition
Packing, Gaskets, Component Seals and O-rings	These would not necessarily be called out explicitly in the scoping and screening. Instead they would be implicitly addressed at the component level. The applicant will be able to exclude these utilizing a clear basis such as the example of ASME Section III not being relied upon for pressure boundary.
Structural Sealants	Structural sealants would not necessarily be called out explicitly in the scoping and screening. Instead they would be implicitly addressed at the component level. Structural sealants may perform functions without moving parts or change in configuration and they are not typically replaced. It is expected that the applicant's structural aging management program will address aging management of these items on a plant specific basis.
Oil, Grease and Component Filters	For these commodities, the screening process would be expected to exclude these materials because they are short-lived.
System Filters, Fire Extinguishers, Fire Hoses and Air Packs o	These may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii) in that they are periodically replaced. The application should identify the standards that are relied on for replacement as part of the method description; for example, NFPA standards for fire protection equipment.

4.2 Identification of Aging Effects Requiring Management

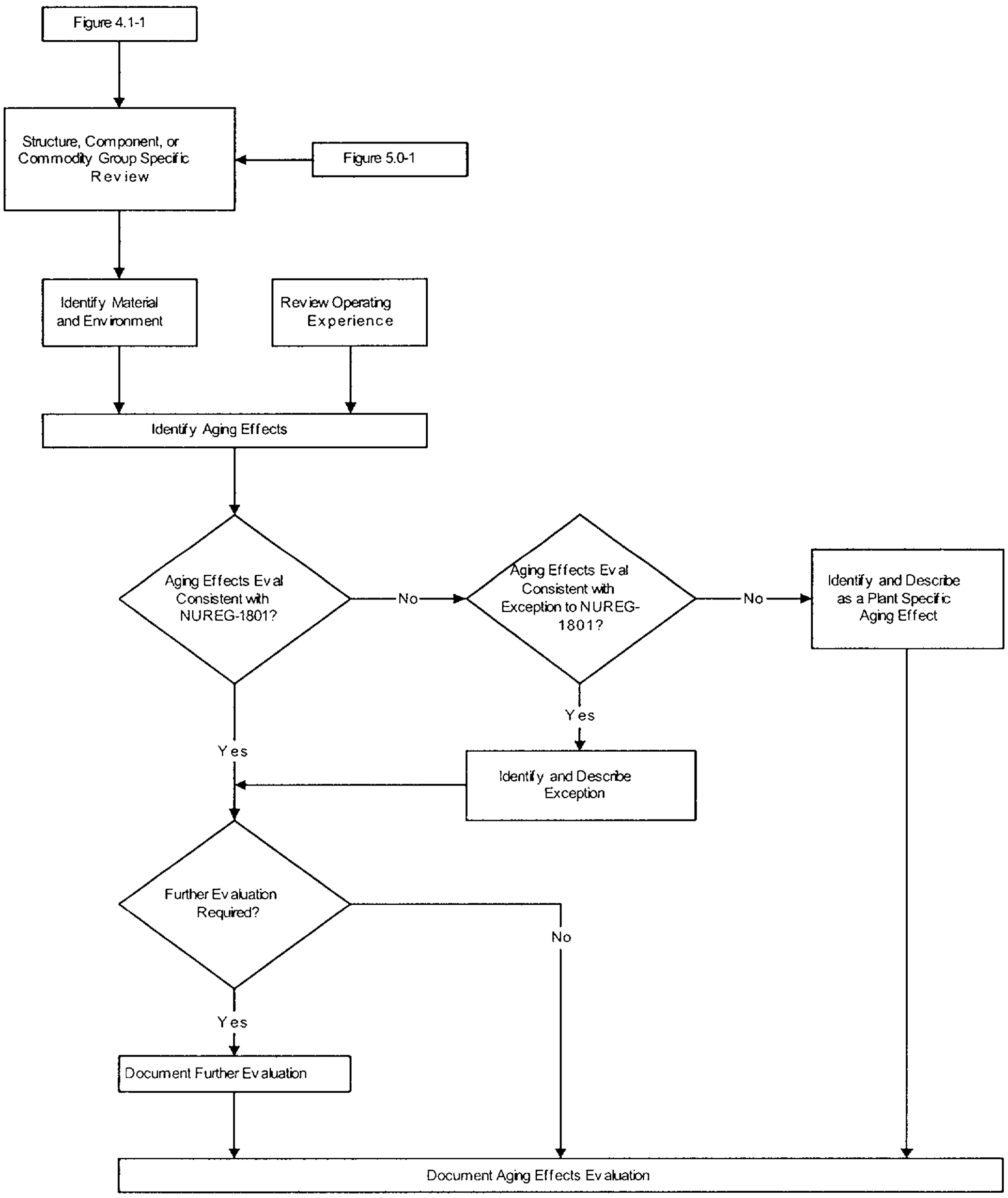
Part 54 Reference

§54.21(a)(3)
(3) For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

This section presents various techniques used to identify aging effects requiring management. However, other techniques may be acceptable provided that the demonstration method required by §54.21(a)(3) is accomplished. Figure 4.2-1 depicts the process to identify aging effects requiring management.

The demonstration required by §54.21(a)(3) is developed by first determining how the structure, component or commodity group performs its intended function(s). Next, the aging effects requiring management are identified. Finally, the applicable plant programs are identified, and the ability to manage the aging effects is reviewed. The assembled information is then used to demonstrate either that the effects of aging will be managed by existing programs so that the structure or component intended function(s) will be maintained for the period of extended operation or that additional aging management activities are necessary.

FIGURE 4.2-1
Identification of Aging Effects Requiring Management



4.2.1 Techniques to Identify Aging Effects

There are various techniques used to identify and assess aging effects. For some structures and components, design margins and/or material properties are known and can be reviewed. In such cases, an analysis may be sufficient to demonstrate that the effects of aging are managed. For other structures and components, performance or maintenance history is available and can be reviewed to assist in demonstrating that the effects of aging are managed. These and other considerations point to the need to determine the appropriate level of review for the type of structure, component, or commodity group and plant-unique conditions.

Assessing the appropriate level of review involves examining information from various investigations and developing a scope statement to describe the depth of review that is needed for the structure, component or commodity group. As appropriate, the assessment should include the following activities:

- Assemble information relative to the structure or component material properties and design margins. If the components are made from different materials or are subject to distinctly different aging effects, a separate review of each may be needed. Because minor differences in chemical content between different alloys may not significantly affect the way in which the materials age, in most cases detailed material specification may not be necessary to identify aging effects.

- Internal and external environments to which components subject to an aging management review (AMR) are exposed should be defined to identify environmental parameters or conditions that are applicable to the environment. A specific environment may be used to bound several environments based on consistency with the specific environmental parameters or conditions.

- Based on material and environment combinations, identify the aging effects potentially affecting the structures' and components' ability to perform their intended function. Various industry documents are available to provide guidance on identification of those aging effects.

- Review the design or material properties to determine if certain aging effects can be shown by analysis not to affect the capability of the structure or component to perform its intended function during the period of extended operation. Of particular interest are parameters such as corrosion allowance, fatigue cycles, loading conditions, fracture toughness, tensile strength, dielectric strength, radiation exposure and environmental exposure.

- Operating experience review is described in section 4.4.

Material-Environment-Stressor Approach

To determine the aging effects requiring management, the applicant should consider and address the materials, environment and stressors that are associated with each structure, component or commodity group under review. In many instances, the proper selection of materials for the operating environment results in few, if any, aging effects requiring management. For example, erosion/corrosion has very little or no aging effects on stainless steel piping. Conversely, carbon steel is subject to erosion/corrosion in a raw water environment. Several industry references identify aging effects based upon specific material-environment combinations. After identification of plant-specific environments and materials, the following industry references could be used as the primary means to identify and evaluate aging effects:

- Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, EPRI 1003056
- Aging Effects for Structures and Structural Components (Structural Tools), EPRI 1002950
- License Renewal Electrical Handbook, EPRI 1003057.

In addition to the consideration of materials, environment, and stressors, the applicant should consider and address the plant-specific CLB, plant and industry operating experience and existing engineering evaluations in order to identify the aging effects requiring management for the structure or component subject to an aging management review. The aging effects requiring management are those that have been identified using the considerations described above, and that adversely affect the structure and component such that the intended function(s) may not be maintained consistent with the CLB for the period of extended operation.

Spaces Approach

The aging management review can also be performed using a "spaces" approach. In the spaces approach, the plant is segregated into areas where common, bounding environmental parameters can be assigned. These areas can be of any size such as a specific area in a room, an entire room, a floor of a building or even all inside areas of an entire building. A bounding environmental parameter, such as temperature, would be the highest average temperature present around the subject components in the defined area.

When used to perform an aging management review of a component or commodity group for a specific environmental stressor, the process would be as follows:

- Identify all component or commodity group materials of construction that have potential aging effects when exposed to the environmental stressor.

- Determine the value of the bounding environmental parameter to which the components in the area to be reviewed are exposed.

- Compare the aging characteristics of the identified materials to the bounding environment and determine if the components will be able to maintain their intended function through the period of extended operation.

Plant-Specific Aging Analysis Based on Loss of Intended Function
By analysis, an applicant may be able to demonstrate that it is not possible for an aging effect to result in a loss of the structure or component's intended function(s) under design basis conditions. The demonstration ultimately should conclude that there is reasonable assurance that the CLB will be maintained for the period of extended operation and therefore that the effects of aging need not be managed. A commitment to an inspection for license renewal, as discussed in Section 4.3, may be needed to verify specific design values, demonstrate that an aging effect is occurring as anticipated, or that an aging effect is not significant. Monitoring industry experience such as the results of inspections for license renewal at other plants may also contribute to the demonstration in these cases.

Use of References Reviewed by the NRC
Plant and generic industry references that provide an aging management review of the same type of structure or component should be reviewed. A search of the public document room indices may identify such reports. References that have been reviewed and approved by the NRC provide an acceptable approach.

In the selected reference, identify the scope, assumptions and limitations affecting the results and conclusions of the analysis. Other characteristics that may need to be identified include the configuration, functions, materials, service conditions and original design parameters (corrosion allowance, loading cycles, etc.) and protective measures (coatings, cathodic protection, etc.) affecting the expected service life of the structure or component.

The identified characteristics of the structure or component in the selected reference should be compared to the plant-specific structure or component. The objective is to demonstrate that the plant characteristics are the same as, or are bounded by, the reference, and therefore, it may be concluded that the selected report is applicable and may be used as a basis for the aging management review of the plant structure or component. Any outlier conditions should be identified and reviewed to show that they are not significant with respect to the results or conclusions of the selected reference. Otherwise, a structure or component-specific aging management review (guideline Section 4.2.1) of the outlier condition should be performed.

4.2.2 Consistency With NUREG-1801 Volume 2 Line Items

Each combination of component type, material, environment and aging effect requiring management should be compared with NUREG-1801 Volume 2 line items to identify consistencies. If there is no corresponding line item in NUREG-1801 Volume 2, the combination is a plant-specific aging evaluation result.

Each applicant should identify how the aging evaluation results align with information in NUREG-1801, Volume 2. This is accomplished through a series of notes identified on Table 4.2-2. All note references with letters are standard notes that will be the same from application to application throughout the industry. Any notes the plant requires that are in addition to the standard notes will be identified by a number and deemed plant-specific.

Table 4.2-2
NUREG-1801 Consistency Notes for Aging Management Review Results

Standard Notes

A. Consistent with NUREG-1801 item for component, material, environment and aging effect. AMP is consistent with NUREG-1801 AMP.

B. Consistent with NUREG-1801 item for component, material, environment and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C. Component is different, but consistent with NUREG-1801 item for material, environment and aging effect. AMP is consistent with NUREG-1801 AMP.

D. Component is different, but consistent with NUREG-1801 item for material, environment and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

E. Consistent with NUREG-1801 item for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

F. Material not in NUREG-1801 for this component.

G. Environment not in NUREG-1801 for this component and material.

H. Aging effect not in NUREG-1801 for this component, material, and environment combination.

I. Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J. Neither the component nor the material and environment combination are evaluated in NUREG-1801.

Plant-Specific Notes

1. Determined on a plant-specific basis.

4.3 Demonstrate That the Effects of Aging Are Managed

The Rule requires an applicant to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

In performing the demonstration, an applicant should consider all programs and activities associated with the structure or component. Plant programs and activities that apply to the structures, components or commodity groups should be reviewed to determine if they include actions to manage the effects of aging.

Aging management programs are generally of four types: prevention, mitigation, condition monitoring and performance monitoring. Prevention programs preclude the aging effect from occurring; for example, coating programs to prevent external corrosion of a tank. Mitigation programs attempt to slow the effects of aging; for example, chemistry programs to mitigate internal corrosion of piping. Condition monitoring programs inspect and examine for the presence of and extent of aging effects; for example, visual inspection of concrete structures for cracking and ultrasonic measurement of pipe wall for erosion-corrosion induced wall thinning. Performance monitoring tests the ability of a structure or component to perform its intended function(s); for example, heat balances on heat exchangers for the heat transfer intended function of the tubes (see Appendix C, Reference 1).

In some instances, more than one type of aging management program may be implemented to ensure that the aging effects are adequately managed to ensure the intended function is maintained in the period of extended operation. For example, managing the internal corrosion of piping may rely on a mitigation program (water chemistry) to minimize susceptibility to corrosion and a condition monitoring program (ultrasonic inspection) to verify that the corrosion is insignificant.

The demonstration is not intended to be a reverification of the structure or component design basis. However, in some cases, verification of a specific design basis parameter may be necessary if that parameter or condition is affected by an aging effect and potentially results in a loss of structure or component intended function. This verification may consist of a physical measurement at susceptible locations or on a sampling basis, as justified, or an evaluation that demonstrates that the aging effect will be at a sufficiently slow rate such that the design basis parameter will not be reduced below a value necessary to assure that the intended function(s) will be maintained during the period of extended operation. For example, a safety-related piping component is designed to have structural integrity under design loads, such as normal, upset, emergency and faulted conditions, in accordance with the plant's CLB. An aging effect that should be evaluated for piping is loss of material due to erosion/corrosion. A loss of material could result in pipe wall thinning below design values rendering the pipe unable to sustain its design loads. However, erosion/corrosion affects piping differently depending on the material of construction. Carbon steel piping may be susceptible to loss of material due to erosion/corrosion, and it would be appropriate to evaluate the pipe wall thickness to verify that this design value remains acceptable. Conversely, stainless steel piping is resistant to loss of material from erosion/corrosion and this aging effect normally would not be significant and, thus, it would not be necessary to evaluate the pipe wall thickness to verify this design value.

To make the required demonstration, an applicant may elect to rely on a single program/activity or a combination of aging management programs/activities. Once the applicant has determined the approach for making the demonstration (i.e., single program/activity, multiple programs/activities) the potential aging management program/activity will be evaluated for the 10 elements noted in Table 4.3-1. Hereafter, aging management program(s), aging management activities or collections of aging management programs and activities used to manage an aging effect will be referred to as an AMP.

Figure 4.3-2 identifies three methods that can be used to review the acceptability of an AMP to manage aging in the period of extended operation. The following sections describe the three methods:

- Section 4.3.1 provides a method to review an AMP to demonstrate that the AMP corresponds to the AMP reviewed and approved in NUREG-1801 Section X or Section XI.

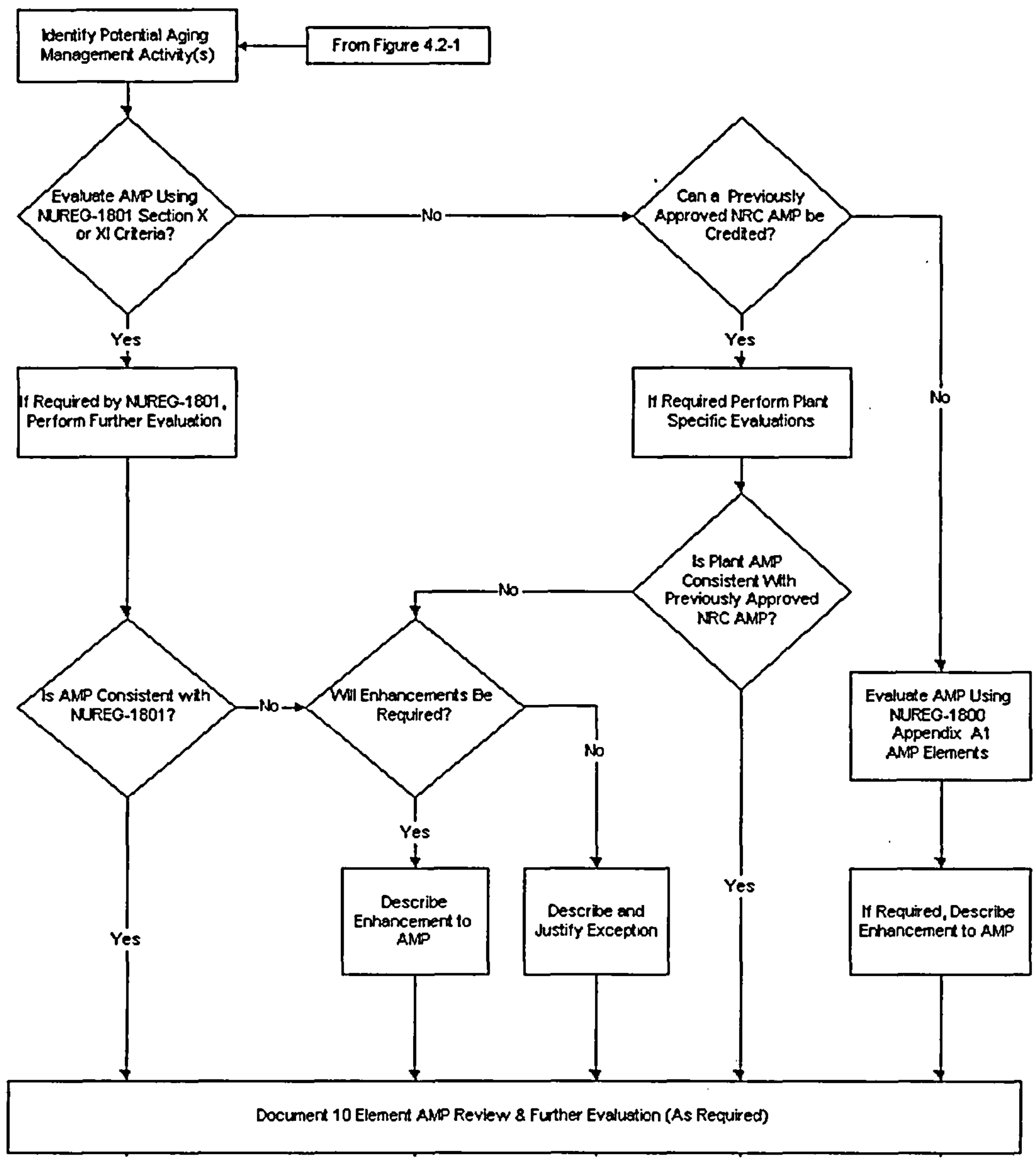
- Section 4.3.2 provides a method to perform a plant-specific evaluation of an AMP that is not described in NUREG-1801.

- Section 4.3.3 provides a method to reference the results of a previous review of an AMP that has been found acceptable by the NRC.

TABLE 4.3-1
Aging Management Activity 10 Program Elements

Element	Description
1. Scope of the activity	Scope of the program/activity should include the specific structures and components subject to an AMR for license renewal.
2. Preventive actions	Preventive actions should mitigate or prevent aging degradation.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure or component intended function(s).
4. Detection of aging effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e. visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and trending	Monitoring and trending should provide predictability of the extent of degradation and provide timely corrective or mitigating actions.
6. Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all current licensing basis design conditions during the period of extended operation.
7. Corrective actions	Corrective actions, including root cause determination and prevention recurrence, should be timely.
8. Confirmation processes	Confirmation processes should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative controls	Administrative controls should provide a formal review and approval process.
10. Operating experience	Operating experience of the aging management activity, including past corrective actions resulting in program enhancements or additional programs or activities, should provide objective evidence to ensure that the effects of aging will be adequately managed so that the intended functions of the structure or component will be maintained during the period of extended operation.

FIGURE 4.3-2
Aging Management Program Review



4.3.1 Aging Management Program Review Using NUREG-1801

The AMP should be reviewed to confirm that it is "consistent with" each of the 10 elements of the generic program described in NUREG-1801 Section X or Section XI. NUREG-1801 documents the NRC staff's basis for determining whether certain programs are adequate to manage aging effects without change or should be augmented to manage aging effects. NUREG-1801 may be referenced in a license renewal application and should be treated in the same manner as an NRC approved topical report. If each of the 10 elements in NUREG-1801 are applicable and consistent with the proposed AMP, the NRC should find that the reference to the NUREG-1801 AMP is acceptable and no further review is required.

Note that NUREG-1801 identifies one acceptable way to manage aging effects. Alternative methods to manage aging may be proposed in the license renewal application. Although the use of NUREG-1801 is not required, its use should facilitate timely, uniform review by the NRC.

If a NUREG-1801 AMP is selected to manage aging, the AMP review should demonstrate consistency of the plant-specific AMP elements with the NUREG-1801 AMP elements. Some engineering judgment may be used in determining that an AMP is "consistent with" NUREG-1801. When there is some expectation that the NRC staff may not come to the same determination with respect to the same program element, the differences should be identified and documented. Any exceptions of the plant AMP to the NUREG-1801 AMP elements should be described and justified. The justification may use an analysis, propose an alternate technique or provide other considerations to confirm that the exception when considered in conjunction with the remainder of the 10 elements would demonstrate that the effects of aging will be adequately managed.

Certain line items in NUREG-1801 mechanical, electrical and structural sections identify AMPs that require further evaluation to augment the specified NUREG-1801 AMP. When required, NUREG-1801 further evaluations must be documented and their conclusions presented in conjunction with the results of the NUREG-1801 AMP evaluation for NRC review.

Enhancements

There may be an AMP where all the NUREG-1801 AMP recommendations cannot be satisfied without appropriate enhancements to the AMP or preparation of a new AMP may be needed. Enhancements are revisions or additions to existing aging management program(s) that will be committed for implementation prior to the period of extended operation. Enhancements may expand, but not reduce the scope of the AMP. Enhancements may include, but are not limited to, verification of specific design values by inspection(s), adding steps to an AMP for specific aging effects, changing the frequency of the required task, adding specific aging effects mitigation procedures, or changing the record-keeping requirements. The factors that should be considered when selecting an appropriate enhancement include:

- The risk significance of the structure or component
- The nature of the aging effect (i.e., is it readily apparent/easily detected?)
- The feasibility of repair/replacement of the affected component or structure
- The compatibility/adaptability of existing programs to detect and manage the aging effect(s)
- The existence of technology to detect and manage the aging effect(s)
- The estimated cost, personnel radiation exposure and impact on normally scheduled outage duration for determining the enhancement.

If existing AMPs, with or without enhancements, are not adequate for managing the effects of aging, new programs or other actions shall be developed as appropriate. One action an applicant could use is a one-time inspection as discussed in NUREG-1801 Section XI.M32. It is possible that an applicant is already performing a relevant inspection or has previously performed an inspection that produced appropriate data for license renewal. Other actions for consideration are refurbishment[3] or replacement.

Quality Assurance and Administrative Controls

Existing 10 CFR 50 Appendix B Quality Assurance programs may be used to generically address the AMP elements of corrective actions, confirmation process and administrative controls for safety related structures, systems and components within the scope of license renewal. For non-safety related structures and components subject to an aging management review, the existing 10 CFR 50 Appendix B Quality Assurance program may be used to address the AMP elements of corrective actions, confirmation process and administrative controls. Alternative means to address the elements of corrective actions, confirmation process and administrative controls for managing aging of non-safety related structures and components that are subject to aging management review may be used but should be consistent with the guidance in NUREG-1800 Appendix A.1 for the applicable elements.

[3] Refurbishment, for purposes of this guideline, means planned actions, short of full replacement, to provide reasonable assurance that the effects of aging are adequately managed such that the intended functions are maintained in accordance with the CLB for the period of extended operation.

4.3.2 Plant-Specific Aging Management Program Review

NUREG-1801 identifies acceptable aging management programs to manage aging effects. Alternative (plant-specific) methods to manage aging may be proposed in the license renewal application. Plant-specific AMPs should be described in terms of the 10 program elements noted in Figure 4.3-1 and the guidance in NUREG-1800 Appendix A.1 "Aging Management Review – Generic (Branch Technical Position RLSB-1)."

The following should be considered when performing a plant-specific AMP review:

- Parameters monitored/inspected: This attribute should include observable parameters or indicators to be monitored or inspected for each aging effect managed. The observable parameters should be linked to the degradation of the structure or component intended functions in the period of extended operation.

- The plant-specific aging management review should either (1) identify an aging management program that detects the effects of aging before the structure or component would lose the ability to perform its intended function, or (2) demonstrate that the structure or component intended function will be maintained during the period of extended operation without the need for an aging management program.

- When an inspection is necessary, sampling may be used to evaluate a group of structures or components. If sampling is used, the program description should describe and justify the methods used for selecting the population and the sample size. A sample consists of one or more structures or components drawn from the scope. The applicant must determine a sample size that is adequate to provide reasonable assurance that the effects of aging on the structure or component population will not prevent the performance of intended functions during the period of extended operation. The size of the sample should include consideration of the specific aging effect, location, existing technical information, materials of construction, service environment, previous failure history, etc. The sample should be biased toward locations most susceptible to the specific aging effect of concern. The results of the inspection also should be evaluated to assess whether the sample size is adequate or if it needs to be expanded.

- An inspection for license renewal may be performed prior to submittal of the license renewal application. The license renewal application may include a commitment to perform an inspection prior to the commencement of the period of extended operation. There also may be justification for performing the inspection during the period of extended operation.

- AMP elements of corrective actions, confirmation process and administrative controls were previously addressed in the "Quality Assurance and Administrative Controls" portion of section 4.3.1.

4.3.3 Use of AMP Previously Approved by NRC

Industry references (e.g. Owners Group Topical Reports, BWRVIP, etc.) that have been approved and reviewed by the NRC can be used to demonstrate that the affects of aging will be managed. The selected reference should also be used to identify the aging effects requiring management and confirm that the assumptions and basis used for determining the aging effects are applicable to the plant. To do this, a review of the plant operating and maintenance history should be performed to confirm that all aging effects apply. Adjustments to the referenced aging effects due to plant-specific conditions may be required. The results may be factored into the description of the aging effects.

The selected reference should be used to identify the programs and features of the programs credited in the review. The comparable plant programs should be identified, and their features should be compared to the programs in the selected reference. Any differences should be identified, and it should be justified that conclusions of the selected reference still apply. The justification may be based on plant-unique features, plant operating and maintenance history, and/or industry developments since the selected reference was issued and reviewed by the NRC. Any plant-specific evaluations required by the reference should be performed.

Any enhancements to current programs or new programs that are cited in the selected reference should be identified. The enhancement(s) that will be implemented for the plant structure or component should be described.

4.4 Operating Experience Review

Industry and plant-specific operating experience requires review to identify aging effects requiring management that are not identified by the industry guidance documents (such as EPRI tools) and to confirm the effectiveness of aging management programs.

Operating Experience – Aging Effects Requiring Management
A plant-specific operating experience review should assess the operating and maintenance history. A review of the prior five to ten years of operating and maintenance history should be sufficient. The results of the review should confirm consistency with documented industry operating experience. Differences with previously documented industry experience such as new aging effects or lack of aging effects allow consideration of plant-specific aging management requirements.

Operating Experience With Aging Management Programs
Plant-specific operating experience with existing programs should be considered. The operating experience of aging management programs, including past corrective actions resulting in program enhancements or additional programs, should be considered. The review should provide objective evidence to support the conclusion that the effects of aging will be managed so that the intended function(s) will be maintained during the extended period of operation. Guidance for reviewing industry operating experience is presented in BTP RLSB-1 in Appendix A.1 of the Branch Technical Positions in NUREG-1800.

Industry Operating Experience
Industry operating experience and its applicability should be assessed to determine whether it changes plant-specific determinations. NUREG-1801 is based upon industry operating experience prior to its date of issue. Operating experience after the issue date of NUERG-1801 should be evaluated and documented as part of the aging management review. In particular, generic communications such as a bulletin or an information notice should be evaluated for impact upon the AMP. The evaluation should check for new aging effects or a new component or location experiencing an already identified aging effect.

4.5 Documenting the Integrated Plant Assessment

Section 54.37(a) of the Rule requires applicants to retain in an auditable and retrievable form all information and documentation required by, or otherwise necessary to document compliance with, the provisions of the Rule.

The results of the IPA should be documented in a format consistent with other plant documentation practices. The information may be maintained in hard-copy or electronic format. It may be appropriate to incorporate the information into an existing plant database if available. The applicant should use the quality assurance program in effect at the plant when documenting the results of the IPA.

4.5.1 Documenting the Identification of Structures and Components Subject to an Aging Management Review

The information to be documented and retained by the applicant should include:

- An identification and listing of structures and components subject to an aging management review and the intended functions
- A description and justification of the methods used to determine the structures and components that are subject to an aging management review
- The information sources used to accomplish the above, and any discussion needed to clarify their use.

The information documented and retained by the applicant will form the bases of the information contained in the application as further discussed in Section 6.

4.5.2 Documenting the Aging Management Review

The information to be documented by the applicant should include:

- An identification of the aging effects requiring management
- An identification of the specific programs or activities that will manage the effects of aging for each structure, component or commodity group listed
- A description of how the programs and activities will manage the effects of aging
- A discussion of how the determinations were made
- A list of substantiating references and source documents

- A discussion of any assumptions or special conditions used in applying or interpreting the source documents

- A description of inspection programs for license renewal.

The information documented and retained by the applicant will form the bases of the information contained in the license renewal application as further discussed in Section 6.

5 TIME-LIMITED AGING ANALYSES INCLUDING EXEMPTIONS

The Rule requires time-limited aging analyses (TLAA) be evaluated. It is intended that TLAAs will capture certain plant-specific aging analyses that are explicitly based on the current operating term of the plant. In addition, the Rule requires exemptions, based on TLAAs, to be identified and analyzed to justify continuation into the period of extended operation. (Figure 5.0-1 outlines the process for evaluating TLAAs and exemptions.)

5.1 Time-Limited Aging Analyses

Part 54 Reference

§54.3

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Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

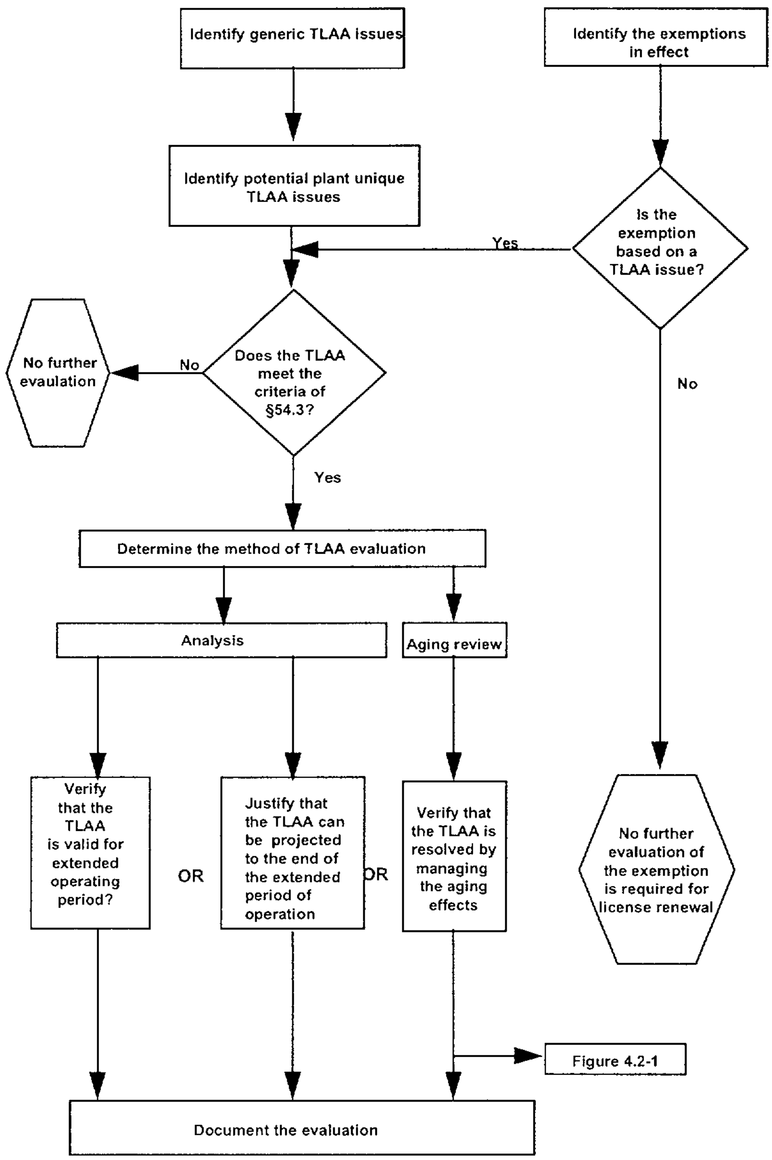
(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);
(2) Consider the effects of aging;
(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
(4) Were determined to be relevant by the licensee in making a safety determination;
(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and
(6) Are contained or incorporated by reference in the CLB.

§54.21(c)(1)

(1) A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that -

(i) The analyses remain valid for the period of extended operation;
(ii) The analyses have been projected to the end of the period of extended operation; or
(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

FIGURE 5.0-1
EVALUATION OF TLAAs AND EXEMPTIONS [§ 54.21(c)]



The applicant must identify the plant-specific TLAA by applying the six criteria delineated in §54.3. The criteria may be applied in any order depending on plant-specific document search capabilities. Guidance for applying the six criteria is provided below.

1. Involve systems, structures and components within the scope of license renewal as delineated in §54.4(a). The system, structure and component scoping step of the integrated plant assessment (Section 3.0) should be performed prior to or concurrent with the TLAA identification.

2. Consider the effects of aging. The effects of aging include but are not limited to loss of material, loss of toughness, loss of prestress, settlement, cracking and loss of dielectric properties.

3. Involve time-limited assumptions defined by the current operating term, for example 40 years. The defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. The assertion should be supported by calculations or other analyses that explicitly include a time limit.

4. Were determined to be relevant by the licensee in making a safety determination. Relevancy is a determination that the licensee must make based on a review of the information available. A calculation or analysis is relevant if it can be shown to have direct bearing on the action taken as a result of the analysis performed. Analyses are also relevant if they provide the basis for the licensee's safety determination and, in the absence of the analyses, the licensee may have reached a different safety conclusion.

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure or component to perform its intended functions as delineated in §54.4(b). As stated in the first criterion, the intended functions must be identified prior to or concurrent with the TLAA identification. Analyses that do not affect the intended functions of the system, structure, or components are not TLAAs.

6. Are contained or incorporated by reference in the current licensing basis (CLB). Plant-specific documents contained or incorporated by reference in the CLB include the FSAR, SERs, Technical Specifications, the fire protection plan/hazards analyses, correspondence to and from the NRC, QA plan, topical reports included as reference to the FSAR or correspondence to the NRC. Calculations and analyses that are not in the CLB or not incorporated by reference are not TLAAs. When the code of record is mentioned in the FSAR, for particular groups of structures or components, referenced material includes all calculations required by that code of record for those structures and components.

All six criteria must be satisfied to conclude that a calculation or analysis is a TLAA. As an aide to applicants, Table 5.1-1 provides examples of how the six criteria may be applied and Table 5.1-2 lists potential TLAA that have been identified from the industry's review of plant-specific CLB documents, various codes, standards and regulatory documents. The table also identifies TLAA considerations that are specifically identified in NUREG-1800 section 4.

TLAAs that need to be addressed are not necessarily those analyses that have been previously reviewed or approved by the NRC. The following examples illustrate TLAAs that need to be addressed and were not previously reviewed and approved by the NRC:

- The FSAR states that the design complies with a certain national code and standard. A review of the code and standard reveals that an analysis or calculation is required. Some of these calculations or analyses will be TLAAs. The actual calculation was performed by the licensee to meet code and standard requirements. The specific calculation was not referenced in the FSAR. The NRC has not reviewed the calculation.

- In response to a generic letter, a licensee submitted a letter to the NRC committing to perform a TLAA that would address the concern in the generic letter. The NRC had not documented a review of the licensee's response and had not reviewed the actual analysis.

The following examples illustrate analyses that are not TLAAs and need not be addressed under 10CFR54.21(c):

- Population projections

- Cost-benefit analysis for plant modifications

- Analysis with time-limited assumptions defined short of the current operating term of the plant; for example, an analysis for a component based on a service life that would not reach the end of the current operating term.

Identified plant-specific TLAAs must be demonstrated acceptable in accordance with §54.21(c)(1) of the Rule. One approach is to verify that the analysis remains valid for the period of extended operation. Guidance for this approach is provided under Section 5.1.1. Another approach is to verify that the analysis can be projected to the end of the period of extended operation. Guidance for this approach is provided in Section 5.1.2. A third approach is to show that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Guidance for this approach is provided in Section 5.1.3.

5.1.1 Verify That the TLAA Is Valid for the Period of Extended Operation

The TLAAs are based on the current operating term (e.g., 40 years). Therefore, the approach outlined in this section may not be applied for the extended operating term, and one of the other approaches (see Sections 5.1.2 and 5.1.3) should be utilized. However, there may be cases where the original analysis or efforts to address new issues during plant operation have resulted in an analysis that can be demonstrated to remain valid for the period of extended operation. A structure or component may have been qualified for at least 40 years. A detailed review of the analysis may demonstrate that the qualification is valid for the period of extended operation and no reanalysis is required. An acceptable approach for verifying that the TLAA remains valid is described in the following paragraphs.

The TLAA issue should be described with respect to the objective(s) of the analysis, conditions and assumptions used in the analysis, acceptance criteria, aging effects requiring management and intended function(s). It should be demonstrated that (1) the conditions and assumptions used in the analysis already address the aging effect(s) requiring management for the period of extended operation, and (2) acceptance criteria are maintained to provide reasonable assurance that the intended function(s) is maintained.

Any actions and an associated implementation plan for reconciling the affected TLAA source documents should be identified.

5.1.2 Justifying the TLAA Can Be Projected to the End of the Period of Extended Operation

The current TLAA may not be valid for the period of extended operation; however, it may be possible to revise the TLAA by recognizing and reevaluating any conservative conditions and assumptions. Examples include relaxing overly conservative assumptions in the original analysis, using new or refined analytical techniques and/or performing the analysis using a 60-year life. The TLAA may then be shown to be valid for the period of extended operation.

5.1.3 Verify That the TLAA Is Resolved by Managing the Aging Effects

The structure(s) or component(s) associated with the TLAA should be identified. The TLAA should be described with respect to the objectives of the analysis, conditions and assumptions used in the analysis, acceptance criteria, aging effect(s) requiring management and intended function(s). The guidance provided in Section 4.2 may be used to demonstrate that the effects of aging on the intended function are adequately managed for the period of extended operation. Also, the monitoring of the aging effect analyzed in the TLAA may include future inspection/examination to detect the aging effect. See NUREG-1801 section X for three programs the NRC has evaluated.

5.1.4 Timing for Evaluation of TLAA

The evaluation of TLAAs could be completed and submitted at the time of renewal application. However, an applicant may defer the completion of the evaluation of TLAAs to a time after the issuance of the renewal license.

When an applicant elects to defer completing the evaluation of a TLAA at the time of renewal application, the applicant should submit the following details in the renewal application to support a conclusion that the effects of aging addressed by that TLAA will be managed for a specific structure or component:

- Details concerning the method that will be used for TLAA evaluation,
- Acceptance criteria that will be used to judge the adequacy of the structure or component, consistent with the CLB, when the TLAA evaluation or analysis is performed,
- Corrective actions that the applicant could perform to provide reasonable assurance that the component in question will perform its intended function when called upon or will not be outside its design basis established by the plant's CLB, and
- Identification of when the TLAA evaluation will be completed to ensure that the necessary evaluation will be performed before the structure or component in question would not be able to perform its intended functions established by the CLB.

TABLE 5.1-1
DISPOSITION OF POTENTIAL TLAAs AND BASIS FOR DISPOSITION

EXAMPLE	DISPOSITION
NRC correspondence requests a utility to justify that unacceptable cumulative wear did not occur during the design life of control rods.	This example does not qualify as a TLAA because the design life of control rods is less than 40 years. Therefore does not meet criterion (3) of the TLAA definition in § 54.3.
Maximum wind speed of 100 mph is expected to occur once per 50 years.	This is not a TLAA. Does not involve an aging effect.
Correspondence from the utility to the NRC states that the membrane on the containment basemat is certified by the vendor to last for 40 years.	This example does not meet criterion (4) of the TLAA definition in § 54.3 and therefore is not considered a TLAA. The membrane was not credited in any safety evaluation.
Fatigue usage factor for the pressurizer surge line was determined not to be an issue for the current license period in response to NRC Bulletin 88-11.	This example is a TLAA because it meets all six criteria in the definition of TLAA in § 54.3. The utility's fatigue design basis relies on assumptions related to 40 year operating life for this component. Plant specific data could be used but is more difficult due to thermal stratification.
Containment tendon liftoff forces are calculated for the 40 year life of the plant. This data is used during Technical Specification surveillance for comparing measured to predicted liftoff forces.	This example is a TLAA because it meets all six criteria of the TLAA definition in § 54.3. The liftoff force curves are limited to 40 year values currently and are needed to perform a required Technical Specification surveillance.

TABLE 5.1-2
POTENTIAL TLAAs

TLAA	NUREG-1800 TLAA Considerations
Reactor Vessel Neutron Embrittlement	- Upper-Shelf Energy - Pressurized Thermal Shock (PWRs) - Pressure-Temperature (P-T) Limits - Elimination of Circumferential Weld Inspection (for BWRs) - Axial Welds (for BWRs)
Metal Fatigue Analysis	- ASME Section III, Class 1 - ANSI B31.1 - Other Evaluations Based on CUF - ASME Section III, Class 2 and 3
Environmental Qualification of Electrical Equipment	- DOR Guidelines - NUREG-0588, Category II (IEEE Std 323-1971) - NUREG-0588, Category I (IEEE Std 323-1974) - GSI 168
Concrete Containment Tendon Prestress	- Concrete Containment Tendon Prestress Analysis
Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis	- ASME Section III, MC or Class 1 - Other Evaluations Based on CUF
Other Plant-Specific TLAAs	- See Note

Note: NUREG-1800 provides general guidance for plant-specific TLAAs. Some examples of plant-specific TLAAs identified in previous license renewal applications include:

- In-service flaw growth analyses that demonstrate structure stability for 40 years
- Containment penetration pressurization cycles
- Fatigue analysis of polar crane (Crane cycle load limits).
- Reactor Coolant Pump Fly Wheel
- Leak-Before-Break Analysis
- Service Water Intake Structure Settlement
- CE-half-nozzle design and mechanical nozzle seal assemblies

5.2 Exemptions

Part 54 Reference

§54.21(c)(2)

(2) A list must be provided of all plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in §54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

Section 54.21(c)(2) of the Rule requires that a list of all exemptions granted under 10 CFR 50.12 that are in effect and based on a TLAA be provided along with the evaluation of time-limited aging analyses.

Identification of an exemption may require the review of correspondence between the NRC and the plant. Many plants have licensing commitment tracking systems or databases of information on licensing documents. As an alternate method or as verification to the search, the NRC docket file in the Public Document Room may be utilized to search for licensing correspondence and, thus, exemptions granted.

It should be determined that the exemption granted pursuant to 10 CFR 50.12 will be in effect during the period of extended operation, involves a system, structure or component within the scope of the Rule and involves a time-limited aging analysis issue. If all of these conditions apply, then an evaluation of the exemption must be performed. The TLAA within the exemption is evaluated using the guidance in Section 5.1

The scope of the exemption, the analysis that forms the basis for the exemption, and the affected structure(s) or component(s) and/or the time-limited aging analysis should be identified. The analysis that forms the basis for the exemption may have been identified during the evaluation of the TLAAs.

The exemption should be evaluated to determine its effect on the capability of the associated plant programs to detect or mitigate the effects of aging or on the conditions and assumptions used in the time-limited aging analysis for the period of extended operation. The evaluation of the associated TLAA may provide sufficient justification to continue the exemption.

5.3 Documenting the Evaluation of the Time-Limited Aging Analyses and Exemptions

Section 54.37(a) of the Rule requires applicants to retain in an auditable and retrievable form all information and documentation required by, or otherwise necessary to document compliance with, the provisions of the Rule.

The results of the time-limited aging analyses and exemptions evaluation should be documented in a format consistent with other plant documentation practices. The information may be maintained in hard-copy or electronic format. If available and appropriate, the information may be incorporated into an existing plant database. The applicant should use the quality assurance program in effect at the plant when documenting the results of the time-limited aging analyses and exemptions evaluation.

The information to be documented by the applicant should include:

- A list of the time-limited aging analyses and exemptions applicable to the plant
- A description of the evaluation performed or to be performed on each plant-specific TLAA and exemption
- A general discussion of how the determinations were made
- A list of substantiating references and source documents
- A discussion of any assumptions or special conditions used in applying or interpreting the source documents.

The information documented and retained by the applicant will form the bases of the information contained in the application as further discussed in Chapter 6.

6 LICENSE RENEWAL APPLICATION FORMAT AND CONTENT

The standard license renewal application format is presented in Table 6.2-1. Table 6.2-2 provides guidance for preparing the standard license renewal application. Contents of the application are general information required by §54.17 and §54.19 and technical information required by §54.21, §54.22 and §54.23.

6.1 General Information

The renewal application contains the technical information that the NRC staff will review to determine if the effects of aging on long-lived passive structures and components are being managed such that the associated intended function(s) is maintained consistent with the current licensing basis (CLB) in the period of extended operation. The technical information must be of sufficient detail to allow the NRC to make the finding that there is reasonable assurance that the activities authorized by the renewed license will continue to be in accordance with the CLB (§54.29(b)).

The application should contain clear and concise presentations of the required information. Confusing or ambiguous statements and unnecessarily verbose descriptions do not contribute to expeditious technical review. Claims of adequacy in the aging management review should be supported by technical bases. The level of detail contained in the application should be commensurate with the requirements of the license renewal rule.

The NRC staff reviewers will use NUREG-1800 and NUREG-1801 during their evaluation of the application. An applicant should consider addressing differences from NUREG-1800 in the application. Generally, applicants will find it beneficial to credit many of the NUREG-1801 evaluations of aging management programs. NUREG-1801 provides one way to manage the aging effects. Other programs may be demonstrated to be adequate. Section 4.3 of this guideline identifies three methods that can be used to demonstrate that the effects of aging are managed.
The application is based on the information contained in plant-specific documentation as described in Sections 3.3, 4.3 and 5.3 of this guideline. However, detailed procedures/calculations need not be included in the license renewal application. Once the license is issued the application is a historical licensing document and is not required to be updated.

6.2 Application Format and Content Guidance

This section provides the standard license renewal application format. Table 6.2-1 is the application table of contents. Guidance for preparing the information for each section of the application is provided in Table 6.2-2. Additional guidelines are provided in Appendix D. This format was developed by applicants who planned submittals to NRC in 2003.

TABLE 6.2-1
STANDARD LICENSE RENEWAL APPLICATION FORMAT

1	**ADMINISTRATIVE INFORMATION**
2	**SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS**
2.1	Scoping and Screening Methodology
2.2	Plant Level Scoping Results
2.3	Scoping and Screening Results: Mechanical Systems
2.3.1	Reactor Coolant System
2.3.2	Engineered Safety Features
2.3.3	Auxiliary Systems
2.3.4	Steam and Power Conversion System
2.4	Scoping and Screening Results: Structures
2.5	Scoping and Screening Results: Electrical and Instrumentation and Controls Systems
3	**AGING MANAGEMENT REVIEW RESULTS**
3.1	Aging Management of Reactor Vessel, Internals and Reactor Coolant System
3.2	Aging Management of Engineered Safety Features
3.3	Aging Management of Auxiliary Systems
3.4	Aging Management of Steam and Power Conversion System
3.5	Aging Management of Containments, Structures and Component Supports
3.6	Aging Management of Electrical and Instrumentation and Controls
4	**TIME-LIMITED AGING ANALYSES**
4.1	Identification of TLAAs
4.2	Reactor Vessel Neutron Embrittlement Analysis
4.3	Metal Fatigue Analysis
4.4	Environmental Qualification (EQ) of Electric Equipment
4.5	Concrete Containment Tendon Prestress Analysis

TABLE 6.2-1
STANDARD LICENSE RENEWAL APPLICATION
FORMAT

4.6	Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis
4.7	Other Plant-Specific TLAAs
APPENDICES	
A: FINAL SAFETY ANALYSIS REPORT (FSAR) SUPPLEMENT	
B: AGING MANAGEMENT PROGRAMS AND ACTIVITIES	
C: (OPTIONAL)	
D: TECHNICAL SPECIFICATION CHANGES	
E: ENVIRONMENTAL INFORMATION	

1 ADMINISTRATIVE INFORMATION

The following information, required by §54.17 and §54.19, is consistent with the information contained in the facility's original operating license application as delineated in 10 CFR 50.33(a) through (e), (h) and (i):

1. Name of applicant
2. Address of applicant
3. Description of business or occupation of applicant
4. Organization and management of applicant

Note that the license renewal rule prohibits any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to know is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, from applying for and obtaining a renewed license.

5. Class of License, the use of the facility and the period of time for which the license is sought.
6. Earliest and latest dates for alterations, if proposed
7. Listing of regulatory agencies having jurisdiction and appropriate news publications (if applicable)
8. Conforming changes to the standard indemnity agreement
9. Restricted data agreement

Pursuant to §54.17 (f) and (g): If the application contains Restricted Data or other defense information, it must be prepared in such a manner that all Restricted Data and other defense information are separated from unclassified information in accordance with 10 CFR 50.33(j). As part of its application and in any event prior to the receipt of Restricted Data or the issuance of a renewed license, the applicant shall agree in writing that it will not permit any individual to have access to Restricted Data until an investigation is made and reported to the Commission on the character, association, and loyalty of the individual and the Commission shall have determined that permitting such persons to have access to Restricted Data will not endanger the common defense and security. The agreement of the applicant in this regard is part of the renewed license, whether so stated or not.

The contents specified for the application are the minimum set required by the regulations. Upon issuance of the renewal operating license, this part of the application becomes a historical document with no further revisions.

2 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

Guidance:

- This subsection provides a brief introduction to Section 2. In addition, it contains Table 2-1, "Intended Functions Abbreviations & Definitions," which contains the meanings for the abbreviations used in the screening and aging management review (AMR) results tables to represent the intended functions for components structural members.

2.1 Scoping and Screening Methodology

Guidance:

- Describe and justify the methodology used to determine the systems, structures and components within the scope of license renewal and the structures and components subject to an AMR [.§54.21(a)(2)].
- The scoping and screening method for mechanical, electrical and civil/structural disciplines may vary. In such cases each method should be described and justified.
- Identify the set of plant-specific design basis events, and corresponding set of plant-specific nomenclature, that the applicant relied on, or that form the basis, to determine the scope of systems, structures and components required in §54.4, consistent with the plant's current licensing basis. Presenting this information in a table or matrix may make the NRC's review more efficient.
- To the extent the maintenance rule scoping criteria are the same for the license renewal rule, licensees may use the same methodology.
- An applicant may attempt to make the NRC review of the license renewal application (LRA) more efficient by indicating its position regarding the subject of any Interim Staff Guidance documents under development at the time of application submittal.

2.2 Plant Level Scoping Results

Guidance:

- Provide a list of all the plant systems and structures identifying those that are within the scope of license renewal. For example, a list may contain 135 plant systems and structures, identifying only 37 that are within the scope of license renewal. If the list exists elsewhere, such as in the FSAR, it is acceptable to merely identify that linkage.
- The license renewal rule does not require the identification of all plant systems and structures. However, providing such a list may make the NRC's review more efficient.

2.3 System Scoping and Screening Results: Mechanical Systems

Guidance:

- Empty heading.

TABLE 6.2-2
GUIDANCE FOR PREPARING THE STANDARD
LICENSE RENEWAL APPLICATION FORMAT

2.3.1 Reactor Coolant System *Guidance:* • For each system, provide the following information: system description to the level of detail that it can be used in the Safety Evaluation Report (SER), system intended functions, FSAR references, reference to drawings submitted with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.
2.3.2 Engineered Safety Features *Guidance:* • For each system, provide the following information: system description to the level of detail that it can be used in the SER, system intended functions, FSAR references, reference to drawings submitted with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.
2.3.3 Auxiliary Systems *Guidance:* • For each system, provide the following information: system description to the level of detail that it can be used in the SER, system intended functions, FSAR references, reference to drawings submitted with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.
2.3.4 Steam and Power Conversion System *Guidance:* • For each system, provide the following information: system description to the level of detail that it can be used in the SER, system intended functions, FSAR references, reference to drawings submitted with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.

2.4 Scoping and Screening Results: Structures *Guidance:* • For each structure, including component supports, subject to aging management review, provide the following information: description to the level of detail that it can be used in the SER, intended functions, FSAR references, reference to drawings submitted with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.
2.5 Scoping and Screening Results: Electrical and Instrumentation and Controls Systems *Guidance:* • Identify electrical and instrumentation and control component types subject to an aging management review [Ref. § 54.21(a)(1)]. For each electrical and instrumentation and control component type provide the following information: description to the level of detail that it can be used in the SER, intended functions, FSAR references, reference to drawings submitted (if applicable) with or as part of the application, a table of component types requiring aging management review with their intended functions and a reference to the section 3 tables with the AMR results for the component types [Ref. §54.21(a)(1)]. • Information concerning interface/boundaries and components/commodities can be described in the text or provided in the form of drawings provided as part of the application or under separate cover.
3 AGING MANAGEMENT REVIEW RESULTS *Guidance:* This subsection contains the roadmap for all of section 3. It identifies where the tables are located (with hyperlinks) that identify the internal and external environments for the SSCs that are subject to aging management review. It also identifies where the table of definitions for abbreviations that are used in section 3 is located (along with its hyperlink). In addition, it includes the following two subsections: • Table Description The purpose of section 3 of the LRA is to present the results of the aging management reviews. The table description section of the LRA describes the two tables that have been developed to present the AMR results information. It describes each column and defines the type of information that each column should contain, including level of detail, where appropriate. • Table Usage This section describes how the two tables work together to present all of the needed information to the reviewer.

3.1 Aging Management of Reactor Vessel, Internals and Reactor Coolant System *Guidance* This subsection is further broken into four subsections. • The introduction provides the road map for the remainder of subsection 3.1. It lists the section of the LRA where the Reactor Vessel, Internals, and Reactor Coolant System SSCs are identified (including a hyperlink). It also lists the systems, or portions of systems, that are addressed in this subsection. Finally, it contains Table 3.1.1, which presents the subsystem information, correlated to the data from Volume 1 of NUREG-1801. • The results contain tables that summarize the aging management reviews for the systems. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Reactor Vessel, Internals, and Reactor Coolant System. Finally, it includes all of the Further Evaluation Recommended information associated with the Reactor Vessel, Internals, and Reactor Coolant System. NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation. • The conclusion contains a conclusion statement regarding the ability of the selected aging management programs (AMPs) to manage the effects of aging on the SCs that are subject to aging management review for the Reactor Vessel, Internals, and Reactor Coolant System. • A list of references is provided.
3.2 Aging Management of Engineered Safety Features *Guidance:* This subsection is further broken into four subsections. • The introduction provides the road map for the remainder of subsection 3.2. It lists the section of the LRA where the Engineered Safety Features SSCs are identified (including a hyperlink). It also lists the systems, or portions of systems, that are addressed in this subsection. Finally, it contains Table 3.2.1, which presents the subsystem information, correlated to the data from Volume 1 of NUREG-1801. • The results contain tables that summarize the aging management reviews for the systems. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Engineered Safety Features. Finally, it includes all of the Further Evaluation Recommended information associated with the Engineered Safety Features. NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation. • The conclusion contains a conclusion statement regarding the ability of the selected AMPs to manage the effects of aging on the SCs that are subject to aging management review for the Engineered Safety Features. • A list of references is provided.

3.3 Aging Management of Auxiliary Systems *Guidance:* This subsection is further broken into four subsections. • The introduction provides the road map for the remainder of subsection 3.3. It lists the section of the LRA where the Auxiliary Systems SSCs are identified (including a hyperlink). It also lists the systems, or portions of systems, that are addressed in this subsection. Finally, it contains Table 3.3.1, which presents the subsystem information, correlated to the data from Volume 1 of NUREG-1801. • The results contain tables that summarize the aging management reviews for the systems. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Auxiliary Systems. Finally, it includes all of the Further Evaluation Recommended information associated with the Auxiliary Systems. NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation. • The conclusion contains a conclusion statement regarding the ability of the selected AMPs to manage the effects of aging on the SCs that are subject to aging management review for the Auxiliary Systems. • A list of references is provided.
3.4 Aging Management of Steam and Power Conversion Systems *Guidance:* This subsection is further broken into four subsections. • The introduction provides the road map for the remainder of subsection 3.4. It lists the section of the LRA where the Steam and Power Conversion Systems SSCs are identified (including a hyperlink). It also lists the systems, or portions of systems, that are addressed in this subsection. Finally, it contains Table 3.4.1, which presents the subsystem information, correlated to the data from Volume 1 of NUREG-1801. • The results contain tables that summarize the aging management reviews for the systems. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Steam and Power Conversion Systems. Finally, it includes all of the Further Evaluation Recommended information associated with the Steam and Power Conversion Systems. NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation. • The conclusion contains a conclusion statement regarding the ability of the selected AMPs to manage the effects of aging on the SCs that are subject to aging management review for the Steam and Power Conversion Systems. • A list of references is provided.
3.5 Aging Management of Containments, Structures and Component Supports

Guidance:

This subsection is further broken into four subsections.

- The introduction provides the road map for the remainder of subsection 3.5. It lists the section of the LRA where the Containments, Structures and Component Supports SSCs are identified (including a hyperlink). It also lists the structures or portions of structures, that are addressed in this subsection. Finally, it contains Table 3.5.1, which presents the structure information, correlated to the data from Volume 1 of NUREG-1801.
- The results contain tables that summarize the aging management reviews for the systems. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Containments, Structures and Component Supports. Finally, it includes all of the Further Evaluation Recommended information NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation.
- The conclusion contains a conclusion statement regarding the ability of the selected AMPs to manage the effects of aging on the SCs that are subject to aging management review for the Containments, Structures and Component Supports.
- A list of references is provided.

3.6 Aging Management of Electrical and Instrumentation and Controls

Guidance:

This subsection is further broken into four subsections.

- The introduction provides the road map for the remainder of subsection 3.6. It lists the section of the LRA where the Electrical and Instrumentation and Controls SCs are identified (including a hyperlink). It also lists the component types that are addressed in this subsection. Finally, it contains Table 3.6.1, which presents the component type information, correlated to the data from Volume 1 of NUREG-1801.
- The results contain tables that summarize the aging management reviews for the component types. This subsection also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each component type within the Electrical and Instrumentation and Controls. Finally, it includes all of the Further Evaluation Recommended information associated with the Electrical and Instrumentation and Controls. NUREG-1801 and NUREG-1800 indicate which attributes of the program need to be evaluated by the NRC reviewer. This section provides the plant-specific information required for this evaluation.
- The conclusion contains a conclusion statement regarding the ability of the selected AMPs to manage the effects of aging on the SCs that are subject to aging management review for the Electrical and Instrumentation and Controls.
- A list of references is provided.

4. TIME-LIMITED AGING ANALYSES *Guidance:* • Empty heading or, at most, it could be a one-paragraph introduction for the section. The Standard Review Plan For License Renewal will not provide a section to review this information. • Not all of the TLAAs identified below will apply to all licensees. If a TLAA listed below is not applicable, the applicant need only state that it does not apply. It is not necessary to justify why it does not apply.
4.1 Identification of TLAAs *Guidance:* • The application shall include a list of time-limited aging analyses, as defined by §54.3. The application should include the identification of the affected systems, structures, and components, an explanation of the time dependent aspects of the calculation or analysis, and a discussion of the TLAA's impact on the associated aging effect. The identification of the results of the time-limited aging analysis review, which may be provided in tabular form, may reference the section in the Integrated Plant Assessment - Aging Management Review chapter where more details of the actual review and disposition (as required by §54.21(c)(1)(i)-(iii)) are located. • The application shall include a demonstration that (1) the analyses remain valid for the period of extended operation, (2) the analyses have been (or have been identified and will be [§54.29(a)]) projected to the end of the period of extended operation or (3) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. • The application shall include a list of plant-specific exemptions granted pursuant to §50.12 and in effect that are based on TLAAs as defined in §54.3. The application shall include an evaluation that justifies the continuation of these exemptions for the period of extended operation. • Summary descriptions of the evaluations of TLAAs for the period of extended operation shall be included in the FSAR supplement (Appendix A).
4.2 Reactor Vessel Neutron Embrittlement *Guidance:* • Disposition chosen for each of the identified TLAAs. Also, provide a reference to the summary description of TLAA evaluations in the FSAR supplement (Appendix A). Use hypertext to link to the appropriate location in the appendix for electronic submittals [§54.21(c)(1) and §54.21(d)].
4.3 Metal Fatigue *Guidance:* • Disposition chosen for each of the identified TLAAs. Also, provide a reference to the summary description of TLAA evaluations in the FSAR supplement (Appendix A). Use hypertext to link to the appropriate location in the appendix for electronic submittals [§54.21(c)(1) and §54.21(d)].

4.4 Environmental Qualification (EQ) of Electric Equipment *Guidance:* • Disposition chosen for each of the identified TLAAs. Also, provide a reference to the summary description of TLAA evaluations in the FSAR supplement (Appendix A). Use hypertext to link to the appropriate location in the appendix for electronic submittals [§54.21(c)(1) and §54.21(d)].
4.5 Concrete Containment Tendon Prestress *Guidance:* • Disposition chosen for each of the identified TLAAs. Also, provide a reference to the summary description of TLAA evaluations in the FSAR supplement (Appendix A). Use hypertext to link to the appropriate location in the appendix for electronic submittals [§54.21(c)(1) and §54.21(d)].
4.6 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis *Guidance:* • Disposition chosen for each of the identified TLAAs. Also, provide a reference to the summary description of TLAA evaluations in the FSAR supplement (Appendix A). Use hypertext to link to the appropriate location in the appendix for electronic submittals [§54.21(c)(1) and §54.21(d)].
4.7 Other Plant-Specific TLAAs *Guidance:* • Identify and evaluate any plant-specific TLAAs.
APPENDIX A: FINAL SAFETY ANALYSIS REPORT SUPPLEMENT *Guidance:* • The contents of the FSAR supplement will be based on the technical information provided in the application. Section 54.21(d) of the Rule requires a summary description of the programs and activities for managing the effects of aging for the period of extended operation as determined by the IPA review. A summary description of the evaluation of time-limited aging analyses for the period of extended operation must also be included in the FSAR supplement. • Guidance contained in NEI 98-03, "Guidelines For Updating Final Safety Analysis Reports" and NEI 96-07, "Guidelines For 10 CFR 50.59 Evaluations" should be considered in the preparation of the FSAR supplement. • In some instances, summary descriptions of programs and activities already exist in the plant FSAR. The applicant may choose to incorporate these existing pages of the FSAR by reference or may choose to include them in the application. • The process to review and approve this change to the plant FSAR should be the same as that which the applicant presently utilizes. • Once the renewed license is issued, the material contained in this Appendix A should be incorporated into the FSAR.

APPENDIX B: AGING MANAGEMENT PROGRAMS AND ACTIVITIES *Guidance:* Lists and describes the aging management programs and activities referenced in Section 3. Most applicants will find it beneficial to credit many of NUREG-1801 evaluations of aging management programs. NUREG-1801 provides one way to manage the aging effects. Other programs may be demonstrated to be adequate. A cross-reference should be provided of the plant's program names to applicable NUREG-1801 Ch. X and XI program names. An alphabetical list, as well as a list by NUREG-1801 program numbers, should be provided. • Appendix B of the LRA consists of the following four subsections: 1. The introduction provides an overview of Appendix B and provides general information to be used by the reviewer while navigating through Appendix B. It contains the following subsections: overview, method of discussion, quality assurance and administrative controls, operating experience and aging management programs. 2. The aging management programs section contains a table that identifies the sample plant aging management programs, along with the corresponding NUREG-1801 program number and name. The programs are listed in the program order of the NUREG-1801. The programs that are consistent with NUREG-1801, or are consistent with exceptions, are listed first, followed by the plant-specific programs. 3. The section for TLAA evaluation of aging management programs required by §54.21(c)(1)(iii) addresses programs credited in the evaluation of TLAAs. 4. A list of references is provided. • See Appendix D for substantially more detail than is provided in this table.
APPENDIX C: (OPTIONAL) *Guidance:* • An applicant may use this appendix for any plant-specific information felt to be required for the application that does not fit well anywhere else.
APPENDIX D: TECHNICAL SPECIFICATION CHANGES *Guidance:* • Appendix D includes appropriate technical specification changes prepared and presented in a manner consistent with the way the applicant normally submits proposed technical specification revisions. Justification may be included herein, or may reference other parts of the license renewal application. Appendix D meets the requirements of §54.22. • Once the renewed license is issued, the proposed changes to technical specifications will be incorporated and issued along with the renewal license. The technical specifications are in a living document and should be maintained in accordance with applicable regulations and plant procedures
APPENDIX E: ENVIRONMENTAL INFORMATION *Guidance:* • 10 CFR 51.53(c) requires a renewal applicant to address certain environmental

impacts in a supplement to the plant's Environmental Report. This supplement is provided as Appendix E to the renewal application. • The format and content of Appendix E should be based on Supplement 1 to Regulatory Guide 4.2, Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses. • Once the renewed license is issued, the environmental information contained in Appendix E will be maintained in accordance with applicable regulations and plant procedures.

7 POST-LICENSE RENEWAL APPLICATION SUBMITTAL ACTIVITIES

Post-license renewal application submittal activities include update of the license renewal application information for current licensing basis (CLB) changes, license renewal application appeals and post-license renewal Final Safety Analysis Report (FSAR) updates for newly identified SSCs.

7.1 Update of the License Renewal Application for CLB Changes

Part 54 Reference

§54.21(b)
CLB changes during NRC review of application. Each year following submittal of the license renewal application and at least 3 months before scheduled completion of the NRC review, an amendment to the renewal application must be submitted that identifies any change to the CLB of the facility that materially affects the contents of the license renewal application, including the FSAR supplement.

The Rule requires that the application including the FSAR supplement be updated yearly and at least three months before scheduled completion of the NRC review, to identify any changes to the facility's current licensing basis that materially affect the application. These changes are provided to the NRC in the form of an amendment to the license renewal application. A CLB change materially affects the contents of the application when including information about the change in the amendment would reasonably be expected to cause the NRC to come to a different conclusion about the subject of the change, than if the information were not included.

The amendment to the application, submitted at least three months before the scheduled completion of the NRC review, should include a list, of "high level future commitments" as described in reference 15. The list should be contained in an update to the UFSAR supplement.

The due date for the annual update and the update submitted at least three months before the scheduled completion of the NRC review may occur close together chronologically. The applicant may desire discussing the need for two updates with NRC. In reference 14, NRC set the precedent of requiring only one update in these circumstances. The scheduled completion of the NRC review is the date on the NRC application review schedule that the safety evaluation is due.

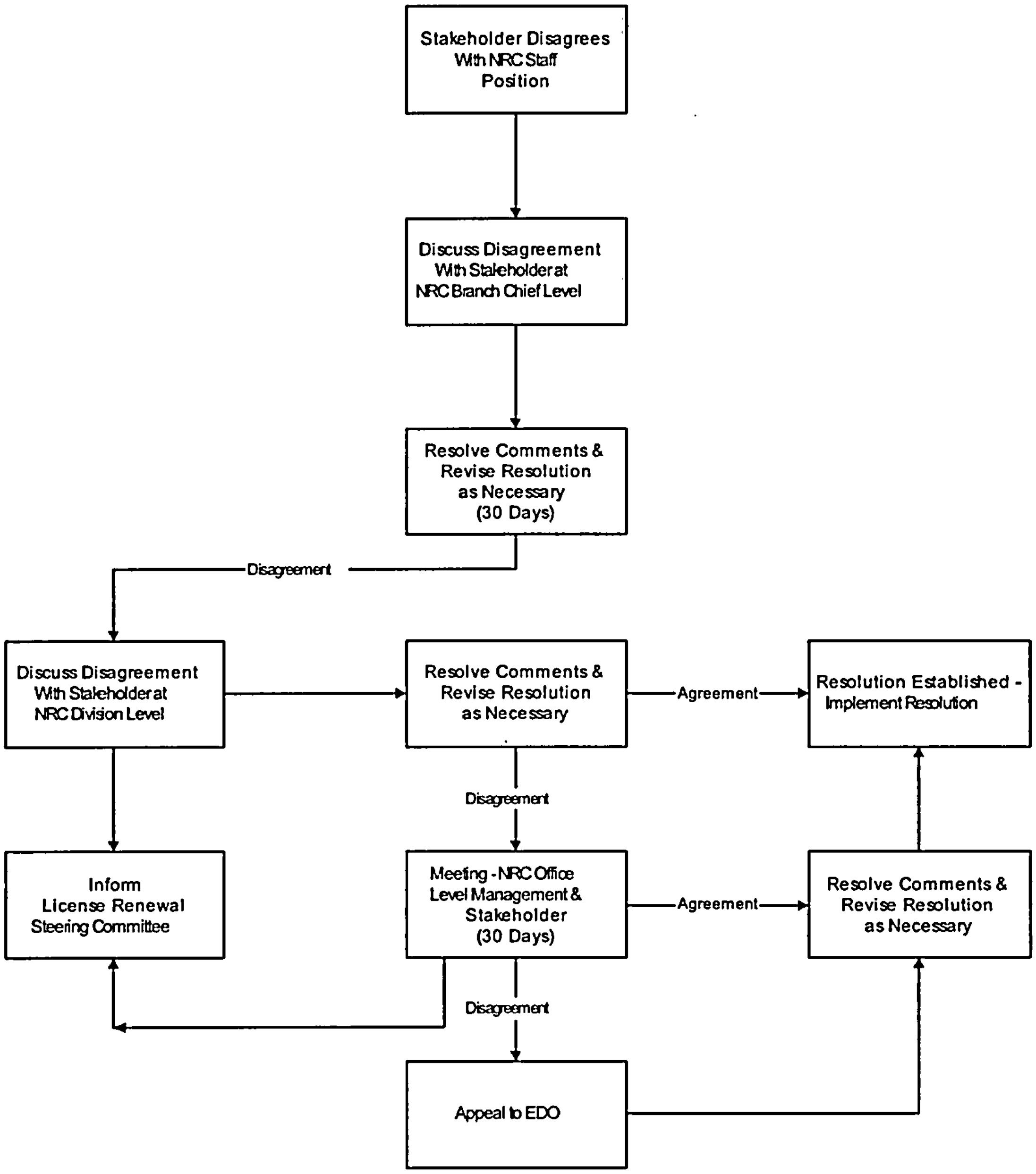
7.2 License Renewal Application Appeals

During review of the license renewal application, any applicant can initiate a formal appeal by a written request to the Program Director, License Renewal & Environmental Impacts Program (PD-RLEP). PD-RLEP will serve as the first-level decision maker in the appeals process. If either party in this first-level appeal wishes to appeal to the division level, such party should submit a written request to the Director, Division of Regulatory Improvement Programs, who will serve as the second-level decision maker. A further appeal can be initiated by a written request to the Director, Office of Nuclear Reactor Regulation, who will serve as the third-level decision maker. The next level of appeal can be initiated by a written request to the Executive Director of Operations, who would serve as the fourth-level decision maker.

The issue being appealed should be clearly defined by a written statement accompanying the request for appeal. The issue statement should have a clearly defined scope and should reference the applicable section(s) of the regulation that provides the requirements for the issue being appealed. Upon receipt of the request for appeal, the PD-RLEP will forward the request to the relevant staff who will review the request and agree that the appeal originator has clearly identified the issue. PD-RLEP will then determine whether the issue is admissible or subject to appeal (i.e., the issue has not previously been decided on appeal). PD-RLEP will provide a written response to the originator, acknowledging receipt of the request, along with the determination of admissibility, and identification of an appeal coordinator, who will provide administrative oversight and support during the appeal process. PD-RLEP's determination regarding the admissibility of the request should include the basis for the determination.

See the License Renewal Appeals Process Flowchart, Figure 7.2-1.

FIGURE 7.2-1
LICENSE RENEWAL APPEALS PROCESS



7.3 Post License Renewal Newly Identified SSCs

Part 54 Reference

§54.37(b)
After the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with 54.21. This FSAR update must describe how the effects of aging will be managed such that the intended function(s) in 54.4(b) will be effectively maintained during the period of extended operation.

After the renewed license is granted, changes may occur to the plant's design and licensing basis. Newly identified SSCs that would have been subject to aging management review must be evaluated to determine whether there are aging effects that require management.

The FSAR update required by 10 CFR 50.71(e) may need to include a description of the SSCs and a description of how the effects of aging will be managed. The description of how the effects of aging are managed can be a reference to an existing aging management program already described in the FSAR, a description of an existing aging management program not previously credited for license renewal, or a description of a new AMP. The descriptions should be to the same level of detail as exists in the FSAR.

If the licensee identifies existing calculations that would have been time-limited aging analysis, then the licensee must evaluate these calculations to determine how the requirements of 54.21(c) will be met. The demonstration required by 54.21(c) may be done using any of the three options provided by 54.21(c)(1) (i), (ii) or (iii).

If TLAAs are identified for inclusion in the FSAR update required by 10 CFR 50.71(e), then the FSAR update must include a summary description of the evaluation of the TLAA to the same level of detail as exists in the FSAR.

The implementation of this requirement may be accomplished by addition to existing processes for configuration management or it may be accomplished by implementation of new processes specifically to implement the requirements of 10 CFR 54.37(b).
NRC inspection for compliance with this requirement is performed in accordance with Inspection Procedure 71003.

APPENDIX A

10 CFR PART 54
THE LICENSE RENEWAL RULE

PART 54—REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR NUCLEAR POWER PLANTS

GENERAL PROVISIONS

AUTHORITY: Secs. 102, 103, 104, 161, 181, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs 201, 202, 206, 88 Stat. 1242, 1244, as amended (42 U.S.C. 5841, 5842), E.O. 12829, 3 CFR, 1993 Comp., p. 570; E.O. 12958, as amended, 3 CFR, 1995 Comp., p. 333; E.O. 12968, 3 CFR, 1995 Comp., p. 391.

SOURCE: 60 FR 22491, May 8, 1995, unless otherwise noted.

GENERAL PROVISIONS

§ 54.1 Purpose.
This part governs the issuance of renewed operating licenses for nuclear power plants licensed pursuant to Sections 103 or 104b of the Atomic Energy

Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

§ 54.3 Definitions.

(a) As used in this part,

Current licensing basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant-specific design-basis information de-fined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

Integrated plant assessment (IPA) is a licensee assessment that demonstrates that a nuclear power plant facility's structures and components requiring aging management review in accordance with § 54.21(a) for license renewal have been identified and that the effects of aging on the functionality of such structures and components will be managed to maintain the CLB such that there is an acceptable level of safety during the period of extended operation.

Nuclear power plant means a nuclear power facility of a type described in 10 CFR 50.21(b) or 50.22.

Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and

(6) Are contained or incorporated by reference in the CLB.

(b) All other terms in this part have the same meanings as set out in 10 CFR50.2 or Section 11 of the Atomic Energy Act, as applicable.

§ 54.4 Scope.

(a) Plant systems, structures, and components within the scope of this part are—

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions—

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shut-down condition; or

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1) (i), (ii), or (iii) of this section.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the bases for including them within the scope of licensee renewal as specified in paragraphs (a) (1)–(3) of this section.

[60 FR 22491, May 8, 1995, as amended at 61 FR 65175, Dec. 11, 1996; 64 FR 72002, Dec. 23, 1999]

EFFECTIVE DATE NOTE: At 64 FR 72002, Dec. 23, 1999, § 54.4 was amended by revising paragraph (a)(1)(iii), effective Jan. 24, 2000. For the convenience of the user, the superseded text is set forth as follows:

§ 54.4 Scope.

(a) * * *

(1) * * *

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

*   *   *   *   *

§ 54.5 Interpretations.

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the

Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

§ 54.7 Written communications.

All applications, correspondence, re-ports, and other written communications shall be filed in accordance with applicable portions of 10 CFR 50.4.

§ 54.9 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act (44 U.S.C. 3501, et seq.). The NRC may not conduct or sponsor, and a per-son is not required to respond to, a collection of information unless it displays a currently valid OMB control number. OMB has approved the information collection requirements contained in this part under control number 3150–0155.

(b) The approved information collection requirements contained in this part appear in §§ 54.13, 54.17, 54.19, 54.21, 54.22, 54.23, 54.33, and 54.37.

[60 FR 22491, May 8, 1995, as amended at 62 FR 52188, Oct. 6, 1997]

§ 54.11 Public inspection of applications.

Applications and documents submitted to the Commission in connection with renewal applications may be made available for public inspection in accordance with the provisions of the regulations contained in 10 CFR part 2.

§ 54.13 Completeness and accuracy of information.

(a) Information provided to the Commission by an applicant for a renewed license or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant must be complete and accurate in all material respects.

(b) Each applicant shall notify the Commission of information identified by the applicant as having, for the regulated activity, a significant implication for public health and safety or common defense and security. An applicant violates this paragraph only if the applicant fails to notify the Com-mission of information that the applicant has identified as having a significant implication for public health and safety or common defense and security. Notification must be provided to the Administrator of the appropriate regional office within 2 working days of identifying the information. This requirement is not applicable to information that is already required to be pro-vided to the Commission by other re-porting or updating requirements.

§ 54.15 Specific exemptions.

Exemptions from the requirements of this part may be granted by the Commission in accordance with 10 CFR 50.12.

§ 54.17 Filing of application.

(a) The filing of an application for a renewed license must be in accordance with subpart A of 10 CFR part 2 and 10 CFR 50.4 and 50.30.

(b) Any person who is a citizen, national, or agent of a foreign country, or any corporation, or other entity which the Commission knows or has reason to know is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government, is ineligible to apply for and obtain a renewed license.

(c) An application for a renewed license may not be submitted to the Commission earlier than 20 years be-fore the expiration of the operating licensee currently in effect.

(d) An applicant may combine an application for a renewed license with applications for other kinds of licenses.

(e) An application may incorporate by reference information contained in previous applications for licenses or license amendments, statements, correspondence, or reports filed with the Commission, provided that the references are clear and specific.

(f) If the application contains Restricted Data or other defense information, it must be prepared in such a manner that all Restricted Data and other defense information are separated from unclassified information in accordance with 10 CFR 50.33(j).

(g) As part of its application, and in any event before the receipt of Restricted Data or classified National Security Information or the issuance of a renewed license, the applicant shall agree in writing that it will not permit any individual to have access to or any facility to possess Restricted Data or classified National Security Information until the individual and/or facility has been approved for such access under the provisions of 10 CFR parts 25 and/or 95. The agreement of the applicant in this regard shall be deemed part of the renewed license, whether so stated therein or not.

[60 FR 22491, May 8, 1995, as amended at 62 FR 17690, Apr. 11, 1997]

§ 54.19 Contents of application—general information.

(a) Each application must provide the information specified in 10 CFR 50.33 (a) through (e), (h), and (i). Alter-natively, the application may incorporate by reference other documents that provide the information required by this section.

(b) Each application must include conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license.

§ 54.21 Contents of application—technical information.

Each application must contain the following information:

(a) An integrated plant assessment (IPA). The IPA must—

(1) For those systems, structures, and components within the scope of this part, as delineated in § 54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management re-view shall encompass those structures and components—

(i)That perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

(ii) That are not subject to replacement based on a qualified life or specified time period.

(2) Describe and justify the methods used in paragraph (a)(1) of this section.

(3) For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

(b) CLB changes during NRC review of the application. Each year following submittal of the license renewal application and at least 3 months before scheduled completion of the NRC re-view, an amendment to the renewal application must be submitted that identifies any change to the CLB of the facility that materially affects the con-tents of the license renewal application, including the FSAR supplement.

(c) An evaluation of time-limited aging analyses.

(1) A list of time-limited aging analyses, as defined in § 54.3, must be provided. The applicant shall demonstrate that—

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

(2) A list must be provided of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in § 54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

(d) An FSAR supplement. The FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by paragraphs (a) and (c) of this section, respectively.

§ 54.22 Contents of application—technical specifications.

Each application must include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. The justification for changes or additions to the technical specifications must be contained in the license renewal application.

§ 54.23 Contents of application—environmental information.

Each application must include a supplement to the environmental report that complies with the requirements of subpart A of 10 CFR part 51.

§ 54.25 Report of the Advisory Committee on Reactor Safeguards.

Each renewal application will be referred to the Advisory Committee on Reactor Safeguards for a review and re-port. Any report will be made part of the record of the application and made available to the public, except to the extent that security classification prevents disclosure.

§ 54.27 Hearings.

A notice of an opportunity for a hearing will be published in the FEDERAL REGISTER in accordance with 10 CFR 2.105. In the absence of a request for a hearing filed within 30 days by a person whose interest may be affected, the Commission may issue a renewed operating license without a hearing upon 30-day notice and publication once in the FEDERAL REGISTER of its intent to do so.

§ 54.29 Standards for issuance of a renewed license.

A renewed license may be issued by the Commission up to the full term authorized by §54.31 if the Commission finds that:

(a) Actions have been identified and have been or will be taken with respect to the matters identified in paragraphs (a)(1) and (a)(2) of this section, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the plant's CLB in order to comply with this paragraph are in accord with the Act and the Commission's regulations. These matters are:

(1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1); and

(2) time-limited aging analyses that have been identified to require review under §54.21(c).

(b) Any applicable requirements of subpart A of 10 CFR part 51 have been satisfied.

(c) Any matters raised under § 2.758 have been addressed.

§ 54.30 Matters not subject to a renewal review.

(a) If the reviews required by § 54.21 (a) or (c) show that there is not reason-able assurance during the current licensee term that licensed activities will be conducted in accordance with the CLB, then the licensee shall take measures under its current license, as appropriate, to ensure that the intended function of those systems, structures or components will be maintained in accordance with the CLB throughout the term of its current license.

(b) The licensee's compliance with the obligation under Paragraph (a) of this section to take measures under its current license is not within the scope of the license renewal review.

§ 54.31 Issuance of a renewed license.

(a) A renewed license will be of the class for which the operating license currently in effect was issued.

(b) A renewed license will be issued for a fixed period of time, which is the sum of the additional amount of time beyond the expiration of the operating license (not to exceed 20 years) that is requested in a renewal application plus the remaining number of years on the operating license currently in effect. The term of any renewed license may not exceed 40 years.

(c) A renewed license will become effective immediately upon its issuance, thereby superseding the operating license previously in effect. If a renewed license is subsequently set aside upon further administrative or judicial appeal, the operating license previously in effect will be reinstated unless its term has expired and the renewal application was not filed in a timely manner.

(d) A renewed license may be subsequently renewed in accordance with all applicable requirements.

§ 54.33 Continuation of CLB and conditions of renewed license.

(a) Whether stated therein or not, each renewed license will contain and otherwise be subject to the conditions set forth in 10 CFR 50.54.

(b) Each renewed license will be issued in such form and contain such conditions and limitations, including technical specifications, as the Commission deems appropriate and necessary to help ensure that systems, structures, and components subject to review in accordance with § 54.21 will continue to perform their intended functions for the period of extended operation. In addition, the renewed licensee will be issued in such form and contain such conditions and limitations as the Commission deems appropriate and necessary to help ensure that systems, structures, and components associated with any time-limited aging analyses will continue to per-form their intended functions for the period of extended operation.

(c) Each renewed license will include those conditions to protect the environment that were imposed pursuant to 10 CFR 50.36b and that are part of the CLB for the facility at the time of issuance of the renewed license. These conditions may be supplemented or amended as necessary to protect the environment during the term of the renewed license and will be derived from information contained in the supplement to the environmental report submitted pursuant to 10 CFR part 51, as analyzed and evaluated in the NRC record of decision. The conditions will identify the obligations of the licensee in the environmental area, including, as appropriate, requirements for re-porting and recordkeeping of environ- mental data and any conditions and monitoring requirements for the protection of the nonaquatic environment.

(d) The licensing basis for the renewed license includes the CLB, as de-fined in §54.3(a); the inclusion in the licensing basis of matters such as licensee commitments does not change the legal status of those matters unless specifically so ordered pursuant to paragraphs (b) or (c) of this section.

§ 54.35 Requirements during term of renewed license.

During the term of a renewed license, licensees shall be subject to and shall continue to comply with all Commission regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, and 100, and the appendices to these parts that are applicable to holders of operating licenses.

§ 54.37 Additional records and record-keeping requirements.

(a) The licensee shall retain in an auditable and retrievable form for the term of the renewed operating license all information and documentation required by, or otherwise necessary to document compliance with, the provisions of this part.

(b) After the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with § 54.21. This FSAR update must describe how the effects of aging will be managed such that the intended function(s) in § 54.4(b) will be effectively maintained during the period of extended operation.

§ 54.41 Violations.

(a) The Commission may obtain an injunction or other court order to pre-vent a violation of the provisions of the following acts—

(1) The Atomic Energy Act of 1954, as amended.

(2) Title II of the Energy Reorganization Act of 1974, as amended or

(3) A regulation or order issued pursuant to those acts.

(b) The Commission may obtain a court order for the payment of a civil penalty imposed under Section 234 of the Atomic Energy Act—

(1) For violations of the following—

(i) Sections 53, 57, 62, 63, 81, 82, 101, 103, 104, 107, or 109 of the Atomic Energy Act of 1954, as amended;

(ii) Section 206 of the Energy Reorganization Act;

(iii) Any rule, regulation, or order issued pursuant to the sections specified in paragraph (b)(1)(i) of this section;

(iv) Any term, condition, or limitation of any license issued under the sections specified in paragraph (b)(1)(i) of this section.

(2) For any violation for which a license may be revoked under Section 186 of the Atomic Energy Act of 1954, as amended.

§ 54.43 Criminal penalties.

(a) Section 223 of the Atomic Energy Act of 1954, as amended, provides for criminal sanctions for willful violations of, attempted violation of, or conspiracy to violate, any regulation issued under sections 161b, 161i, or 161o of the Act. For purposes of section 223, all the regulations in part 54 are issued under one or more of sections 161b, 161i, or 161o, except for the sections listed in paragraph (b) of this section.

(b) The regulations in part 54 that are not issued under Sections 161b, 161i, or 161o for the purposes of Section 223 are as follows: §§ 54.1, 54.3, 54.4, 54.5, 54.7, 54.9, 54.11, 54.15, 54.17, 54.19, 54.21, 54.22, 54.23, 54.25, 54.27, 54.29, 54.31, 54.41, and 54.43.

APPENDIX B

TYPICAL STRUCTURE, COMPONENT AND COMMODITY GROUPINGS AND ACTIVE/PASSIVE DETERMINATIONS FOR THE INTEGRATED PLANT ASSESSMENT

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
1	Structures	Category I Structures Note: If a dam is included in this category – see Appendix C, Reference 4 for guidance on an acceptable aging management program.	Yes
2	Structures	Primary Containment Structure	Yes
3	Structures	Intake Structures	Yes
4	Structures	Intake Canal	Yes
5	Structures	Other Non-Category I Structures Within the Scope of License Renewal Note: If a dam is included in this category – see Appendix C, Reference 4 for guidance on an acceptable aging management program	Yes
6	Structures	Equipment Supports and Foundations	Yes
7	Structures	Structural Bellows	Yes
8	Structures	Controlled Leakage Doors	Yes
9	Structures	Penetration Seals	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
10	Structures	Compressible Joints and Seals	Yes
11	Structures	Fuel Pool and Sump Liners	Yes
12	Structures	Concrete Curbs	Yes
13	Structures	Offgas Stack and Flue	Yes
14	Structures	Fire Barriers	Yes
15	Structures	Pipe Whip Restraints and Jet Impingement Shields	Yes
16	Structures	Electrical and Instrumentation and Control Penetration Assemblies	Yes
17	Structures	Instrumentation Racks, Frames, Panels, and Enclosures	Yes
18	Structures	Electrical Panels, Racks, Cabinets, and Other Enclosures	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
19	Structures	Cable Trays and Supports	Yes
20	Structures	Conduit	Yes
21	Structures	Tube Track	Yes
22	Structures	Reactor Vessel Internals	Yes
23	Structures	ASME Class 1 Hangers and Supports	Yes
24	Structures	Non-ASME Class 1 Hangers and Supports	Yes
25	Structures	Snubbers	No
26	Reactor Coolant Pressure Boundary Components (Note: the components of the RCPB are defined by each plant's CLB and site specific documentation.	ASME Class 1 Piping	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
27	Reactor Coolant Pressure Boundary Components	Reactor Vessel	Yes
28	Reactor Coolant Pressure Boundary Components	Reactor Coolant Pumps	Yes (Casing)
29	Reactor Coolant Pressure Boundary Components	Control Rod Drives	No
30	Reactor Coolant Pressure Boundary Components	Control Rod Drive Housing	Yes
31	Reactor Coolant Pressure Boundary Components	Steam Generators	Yes
32	Reactor Coolant Pressure Boundary Components	Pressurizers	Yes
33	Non-Class I Piping Components	Underground Piping	Yes
34	Non-Class I Piping Components	Piping in Low Temperature Demineralized Water Service	Yes
35	Non-Class I Piping Components	Piping in High Temperature Single Phase Service	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
36	Non-Class I Piping Components	Piping in Multiple Phase Service	Yes
37	Non-Class I Piping Components	Service Water Piping	Yes
38	Non-Class I Piping Components	Low Temperature Gas Transport Piping	Yes
39	Non-Class I Piping Components	Stainless Steel Tubing	Yes
40	Non-Class I Piping Components	Instrument Tubing	Yes
41	Non-Class I Piping Components	Expansion Joints	Yes
42	Non-Class I Piping Components	Ductwork	Yes
43	Non-Class I Piping Components	Sprinklers Heads	Yes
44	Non-Class I Piping Components	Miscellaneous Appurtenances (Includes fittings, couplings, reducers, elbows, thermowells, flanges, fasteners, welded attachments, etc.)	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
45	Pumps	ECCS Pumps	Yes (Casing)
46	Pumps	Service Water and Fire Pumps	Yes (Casing)
47	Pumps	Lube Oil and Closed Cooling Water Pumps	Yes (Casing)
48	Pumps	Condensate Pumps	Yes (Casing)
49	Pumps	Borated Water Pumps	Yes (Casing)
50	Pumps	Emergency Service Water Pumps	Yes (Casing)
51	Pumps	Submersible Pumps	Yes (Casing)
52	Turbines	Turbine Pump Drives (excluding pumps)	Yes (Casing)
53	Turbines	Gas Turbines	Yes (Casing)

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
54	Turbines	Controls (Actuator and Overspeed Trip)	No
55	Engines	Fire Pump Diesel Engines	No
56	Emergency Diesel Generators	Emergency Diesel Generators	No
57	Heat Exchangers	Condensers	Yes
58	Heat Exchangers	HVAC Coolers (including housings)	Yes
59	Heat Exchangers	Primary Water System Heat Exchangers	Yes
60	Heat Exchangers	Treated Water System Heat Exchangers	Yes
61	Heat Exchangers	Closed Cooling Water System Heat Exchangers	Yes
62	Heat Exchangers	Lubricating Oil System Heat Exchangers	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
63	Heat Exchangers	Raw Water System Heat Exchangers	Yes
64	Heat Exchangers	Containment Atmospheric System Heat Exchangers	Yes
65	Miscellaneous Process Components	Gland Seal Blower	No
66	Miscellaneous Process Components	Recombiners	The applicant shall identify the intended function and apply the IPA process to determine if the grouping is active or passive.
67	Miscellaneous Process Components	Flexible Connectors	Yes
68	Miscellaneous Process Components	Strainers	Yes
69	Miscellaneous Process Components	Rupture Disks	Yes
70	Miscellaneous Process Components	Steam Traps	Yes
71	Miscellaneous Process Components	Restricting Orifices	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
72	Miscellaneous Process Components	Air Compressor	No
73	Electrical and I&C	Alarm Unit (e.g., fire detection devices)	No
74	Electrical and I&C	Analyzers (e.g., gas analyzers, conductivity analyzers)	No
75	Electrical and I&C	Annunciators (e.g., lights, buzzers, alarms)	No
76	Electrical and I&C	Batteries	No
77	Electrical and I&C	Cables and Connections, Bus, electrical portions of Electrical and I&C Penetration Assemblies, Includes fuse holders outside of cabinets of active electrical SCs (e.g., electrical penetration assembly cables and connections, connectors, electrical splices, terminal blocks, power cables, control cables, instrument cables, insulated cables, communication cables, uninsulated ground conductors, transmission conductors, isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, switchyard bus)	Yes
78	Electrical and I&C	Chargers, Converters, Inverters (e.g., converters-voltage/current, converters-voltage/pneumatic, battery chargers/inverters, motor-generator sets)	No
79	Electrical and I&C	Circuit Breakers (e.g., air circuit breakers, molded case circuit breakers, oil-filled circuit breakers)	No

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
80	Electrical and I&C	Communication Equipment (e.g., telephones, video or audio recording or playback equipment, intercoms, computer terminals, electronic messaging, radios, transmission line traps and other power-line carrier equipment)	No
81	Electrical and I&C	Electric Heaters,	No, Yes for a Pressure Boundary if applicable, See Appendix C Reference 2
82	Electrical and I&C	Heat Tracing	No See Appendix C Reference 2
83	Electrical and I&C	Electrical Controls and Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, fuse holders, etc.) (e.g., main control board, HVAC control board)	No
84	Electrical and I&C	Elements, RTDs, Sensors, Thermocouples, Transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors,watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, var. transducers, vibration transducers, voltage transducers)	No Yes for a Pressure Boundary if applicable
85	Electrical and I&C	Fuses	No See Appendix C Reference 3
86	Electrical and I&C	Generators, Motors (e.g., emergency diesel generators, ECCS and emergency service water pump motors, small motors, motor-generator sets, steam turbine generators, combustion turbine generators, fan motors, pump motors, valve motors, air compressor motors)	No

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
87	Electrical and I&C	High-voltage Insulators (e.g., porcelain switchyard insulators, transmission line insulators)	Yes
88	Electrical and I&C	Surge Arresters (e.g., switchyard surge arresters, lightning arresters, surge suppressers, surge capacitors, protective capacitors)	No
89	Electrical and I&C	Indicators (e.g., differential pressure indicators, pressure indicators, flow indicators, level indicators, speed indicators, temperature indicators, analog indicators, digital indicators, LED bar graph indicators, LCD indicators)	No
90	Electrical and I&C	Isolators (e.g., transformer isolators, optical isolators, isolation relays, isolating transfer diodes)	No
91	Electrical and I&C	Light Bulbs (e.g., indicating lights, emergency lighting, incandescent light bulbs, fluorescent light bulbs)	No See Appendix C Reference 2
92	Electrical and I&C	Loop Controllers (e.g., differential pressure indicating controllers, flow indicating controllers, temperature controllers, controllers, speed controllers, programmable logic controller, single loop digital controller, process controllers, manual loader, selector station, hand/auto station, auto/manual station)	No
93	Electrical and I&C	Meters (e.g., ammeters, volt meters, frequency meters, var meters, watt meters, power factor meters, watt-hour meters)	No
94	Electrical and I&C	Power Supplies	No

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
95	Electrical and I&C	Radiation Monitors (e.g., area radiation monitors, process radiation monitors)	No
96	Electrical and I&C	Recorders (e.g., chart recorders, digital recorders, events recorders)	No
97	Electrical and I&C	Regulators (e.g., voltage regulators)	No
98	Electrical and I&C	Relays (e.g., protective relays, control/logic relays, auxiliary relays)	No
99	Electrical and I&C	Signal Conditioners	No
100	Electrical and I&C	Solenoid Operators	No
101	Electrical and I&C	Solid-State Devices (e.g., transistors, circuit boards, computers)	No
102	Electrical and I&C	Switches (e.g., differential pressure indicating switches, differential pressure switches, pressure indicator switches, pressure switches, flow switches, conductivity switches, level indicating switches, temperature indicating switches, temperature switches, moisture switches, position switches, vibration switches, level switches, control switches, automatic transfer switches, manual transfer switches, manual disconnect switches, current switches, limit switches, knife switches)	No

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
103	Electrical and I&C	Switchgear, Load Centers, Motor Control Centers, Distribution Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, etc.) (e.g., 4.16 kV switchgear, 480V load centers, 480V motor control centers, 250 VDC motor control centers, 6.9 kV switchgear units, 240/125V power distribution panels)	No
104	Electrical and I&C	Transformers (e.g., instrument transformers, load center transformers, small distribution transformers, large power transformers, isolation transformers, coupling capacitor voltage transformers)	No See Appendix C Reference 2
105	Electrical and I&C	Transmitters (e.g., differential pressure transmitters, pressure transmitters, flow transmitters, level transmitters, radiation transmitters, static pressure transmitters)	No
106	Valves	Hydraulic Operated Valves	Yes (Bodies)
107	Valves	Explosive Valves	Yes (Bodies)
108	Valves	Manual Valves	Yes (Bodies)
109	Valves	Small Valves	Yes (Bodies)
110	Valves	Motor-Operated Valves	Yes (Bodies)

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
111	Valves	Air-Operated Valves	Yes (Bodies)
112	Valves	Main Steam Isolation Valves	Yes (Bodies)
113	Valves	Small Relief Valves	Yes (Bodies)
114	Valves	Check Valves	Yes (Bodies)
115	Valves	Safety Relief Valves	Yes (Bodies)
116	Valves	Dampers, louvers, and gravity dampers	Yes (Housings)
117	Tanks	Air Accumulators	Yes
118	Tanks	Discharge Accumulators (Dampers)	Yes
119	Tanks	Boron Acid Storage Tanks	Yes

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUPING	STRUCTURE, COMPONENT, OR COMMODITY GROUPING MEETS 10CFR54.21(a)(1)(i) (YES/NO)
120	Tanks	Above Ground Oil Tanks	Yes
121	Tanks	Underground Oil Tanks	Yes
122	Tanks	Demineralized Water Tanks	Yes
123	Tanks	Neutron Shield Tank	Yes
124	Fans	Ventilation Fans (includes intake fans, exhaust fans, and purge fans)	Yes (Housings)
125	Fans	Other Fans	Yes (Housings)
126	Miscellaneous	Emergency Lighting	No
127	Miscellaneous	Hose Stations	Yes

APPENDIX C

REFERENCES

Appendix C

References

- Reference 1: *LICENSE RENEWAL ISSUE NO. 98-0105, "HEAT EXCHANGERS HEAT TRANSFER FUNCTION,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC dated November 19, 1999

- Reference 2: *DETERMINATION OF AGING MANAGEMENT REVIEW FOR ELECTRICAL COMPONENTS,* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated September 19, 1997

- Reference 3: *LICENSE RENEWAL ISSUE NO. 98-0016, "AGING MANAGEMENT REVIEW OF FUSES,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated April 27, 1999

- Reference 4: *LICENSE RENEWAL ISSUE NO. 98-0100, "CREDITING FERC-REQUIRED INSPECTION AND MAINTENANCE PROGRAMS FOR DAM AGING MANAGEMENT,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated May 5, 1999

- Reference 5: *LICENSE RENEWAL ISSUE NO. 98-0104, "ACCEPTANCE REVIEW OF LICENSE RENEWAL APPLICATIONS,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated February 1, 2000

- Reference 6: *GUIDANCE ON ADDRESING GSI-168 FOR LICENSE RENEWAL,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated June 2, 1998

- Reference 7: *LICENSE RENEWAL ISSUE NO. 98-0051, "EVALUATION OF JURISDICTION OF ASME SECTION XI, SUBSECTIONS IWE AND IWF, FOR LICENSE RENEWAL,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated March 6, 2000

- Reference 8: *LICENSE RENEWAL ISSUE NO. 98-12, "CONSUMABLES,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated March 10, 2000

- Reference 9: *LICENSE RENEWAL ISSUE NO. 98-0013, "DEGRADATION INDUCED HUMAN ACTIVITIES,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated June 5, 1998

- Reference 10: *LICENSE RENEWAL ISSUE NO. 98-0014, "STAFF GUIDANCE FOR LICENSE RENEWAL APPLICATION SUBMITTALS ON TIME-LIMITED AGING ANALYSES FOR ENVIRONMENTAL QUALIFICATION,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated September 24, 1998

- Reference 11: *"GENERIC SAFETY ISSUES RELATED TO LICENSE RENEWAL (TAC NO. M92972),"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated January 29, 1998

- Reference 12: *"LICENSE RENEWAL ISSUE NO. 98-0082, SCOPING GUIDANCE,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated August 5, 1999

- Reference 13: *"LICENSE RENEWAL ISSUE NO. 98-0030 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL COMPONENTS,"* Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated May 19, 2000

- Reference 14: *"ST. LUCIE, UNITS 1 AND 2, EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 54, SECTION 54.21(b) REGARDING SCHEDULE FOR SUBMITTING AMENDMENTS TO THE LICENSE RENEWAL APPLICATION (TAC NOS. MB3406 AND MB3412)",* Letter to J. A. Stall, Florida Power and Light Company from Noel F. Dudley, NRC, dated November 19, 2002

- Reference 15: *"INDUSTRY RESPONSE – CONSOLIDATED LIST OF COMMITMENTS FOR LICENSE RENEWAL, DECEMBER 16, 2002,"* Letter to P.T. Kuo, NRC, from Alan Nelson, NEI, dated February 26, 2003

REFERENCE 1

LICENSE RENEWAL ISSUE NO. 98-0105, "HEAT EXCHANGERS HEAT TRANSFER FUNCTION," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC dated November 19, 1999

November 19, 1999

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0105, "HEAT EXCHANGERS HEAT TRANSFER FUNCTION"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution of the subject issue. The staff found that a clarification should be added to the Standard Review Plan for License Renewal and NEI 95-10. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Hai-Boh Wang at 301-415-2958.

Sincerely,

/Signed/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690

Enclosure: As stated

cc w/encl: See next page

LICENSE RENEWAL ISSUE NO. 98-0105
HEAT EXCHANGERS HEAT TRANSFER FUNCTION

1 BACKGROUND

Section 54.21(a)(1)(i) of Title 10 of the Code of Federal Regulations specifies that heat exchangers are components that are subject to an aging management review and that perform an intended function without moving parts or without a change in configuration or properties.

Section 3.0.III.C of the draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) states, in part: "Performance monitoring programs test the ability of a structure or component to perform its intended function(s), for example, heat balances on heat exchangers for the heat transfer intended function of the tubes."

Experience from the first two renewal applications and industry comments on the generic renewal guidance has demonstrated that, while it is generally understood that the pressure boundary function of the heat exchanger is within the scope of license renewal, some believe that heat exchangers are active with respect to the heat transfer function, and that the heat transfer intended function need not be subject to a separate aging management review.

2. EVALUATION

In 10 CFR 54.21, the following requirement is stated: "Each application must contain the following information: (a) An integrated plant assessment (IPA). The IPA must—

> (1) For those systems, structures, and components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components -
>
> (i) That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to... steam generators... heat exchangers, ventilation ducts... the containment, the containment liner...."

As stated in 10 CFR 54.21(a)(1)(i), heat exchangers perform their intended function(s) without moving parts or without a change in configuration or properties. The staff believes that the Commission intended to include the pressure boundary function and the heat transfer function. The pressure boundary is maintained by the shell and other parts of the heat exchangers. Heat transfer is conducted through the tube wall, which may be made from different materials. Although the cooling fluid is moving and may involve local boiling (a change of state), heat exchangers do not have any moving parts. Therefore, the staff does not believe that the heat transfer function could be reasonably described as "active."

Furthermore, the Statement of Consideration (SOC) (60 FR 22469) states the following:

> "The Commission believes that regardless of the specific aging mechanism, only aging degradation that leads to degraded performance or condition (i.e., detrimental effects) during the period of extended operation is of principal concern for license renewal. Because the detrimental effects of aging are manifested in degraded performance or condition, an appropriate license renewal review would ensure that licensee programs adequately monitor performance or condition in a manner that allows for the timely identification and correction of degraded conditions. The Commission concludes that a shift in focus to managing the detrimental effects of aging for license renewal review is appropriate and will provide reasonable assurance that systems, structures, and components are capable of performing their intended function during the period of extended operation."

This objective can be best achieved by considering both the pressure boundary and heat transfer functions for heat exchangers, because heat transfer is a primary safety function of these components. There may be a unique aging effect associated with different materials in the heat exchanger parts that are associated with the heat transfer function and not the pressure boundary function. The staff would expect that the programs that effectively manage aging effects of the pressure boundary function can, in conjunction with the procedures for monitoring heat exchanger performance, effectively manage aging effects applicable to the heat transfer function.

Heat transfer is also a parameter considered in the design of most of the other safety-related structures and components, but not as a primary safety function like that associated with steam generators and heat exchangers. For example, while the heat capacity of the containment and interior structures is included in the modeling of the pressure and temperature transient for loss-of-coolant accidents, these secondary heat-transfer functions of safety-related structures and components need not be a specific focus of the aging management review for license renewal.

3 RESOLUTION

On the basis of the preceding evaluation, the staff has determined that its proposed position as stated in SRP-LR Section 3.0.III.C is consistent with the rule. However, the clarification of the distinction between the pressure boundary and heat transfer functions, as well and the distinction between the primary and secondary heat transfer functions should be added to the SRP-LR as well as NEI95-10.

REFERENCE 2

DETERMINATION OF AGING MANAGEMENT REVIEW FOR ELECTRICAL COMPONENTS, Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated September 19, 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C 20555-0001

September 19, 1997

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW
Suite 300
Washington, DC 20585

SUBJECT: DETERMINATION OF AGING MANAGEMENT REVIEW FOR ELECTRICAL COMPONENTS

Dear Mr. Walters:

During the Nuclear Regulatory Commission staff's review of the Nuclear Energy Institute's NEI 95-10. "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," the need was identified for guidance on whether selected electrical components are subject to an aging management review. NEI addressed a number of the components in its letter dated December 24, 1996. Consistent with the staff's approach in its February 27. 1997, letter to provide positions on significant issues associated with the license renewal regulatory guide and NEI 95-10, enclosed please find the staff's position on the aging management review requirements for selected electrical components. The recommendations in the enclosed position should be considered when revising NEI 95-10.

Sincerely,

Christopher I. Crimes. Director
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project 690

Enclosure: As stated

cc: w/encl: See next page

Determination of aging management review for electrical components

Issue:

Determining if transformers, fuses, indicating lights9 heat tracing, electric heaters, and recombiners are subject to an aging management review.

NRC staff position:

This issue relates to the guidance provided in the Statements of Consideration (SOC) in which the Commission concluded that an aging management review is required for passive, long-lived structures and components within the scope of the license renewal rule. Appendix B of NE! 95-10 addresses this requirement by identifying typical structure, component, and commodity groupings and a determination as to whether they require an aging management review. Several electrical components, as identified above, were not classified in Appendix B. The rule in §54.21(a)(1), states that "structures and components subject to an aging management review shall encompass those structures and components (i) [t]hat perform an intended function as described in §54.4, without moving parts or without a change In configuration or properties." The SOC uses the term "passive" to represent these characteristics for convenience. The description of "passive" structures and components incorporated into §54.21(a)(1)(i) is used only In conjunction with the IPA review in the license renewal process. The SOC accompanying the renewal rule states: "The Commission has determined that passive structures and components for which aging degradation is not readily monitored are those that perform an intended function without moving parts or a change in configuration or properties." (60 FR 22477). The SOC also states: "[T]he commission has concluded that "a change in configuration or properties should be interpreted to include "a change in state," which is a term sometimes found in the literature relating to "passive."

§54.21(a)(1)(i) excludes a variety of electrical and instrumentation and control (I&C) structures and components from an aging management review for renewal such as motors, diesel generators, air compressors, pressure transmitters, pressure indicators, water level indicators, switchgear, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies. The SOC provides the following discussion as the basis for excluding several electrical and !&C devices from an aging management review: "an electrical relay can change its configuration, and a battery changes its electrolyte properties when discharging" and "a transistor can 'change its state'." The SOC also provides the following discussion as the basis to include electrical cables in an aging management review: "they perform their intended function without moving parts or without a change in configuration or properties and the effects of aging degradation for these components are not readily monitorable." (60 FR 22477)

While §54.21(a)(1)(i) excludes many electrical and I&C components from an aging management review for renewal, It also states that the exclusion is "not limited to" only these components. The staff has considered the aging

Attachment

management review requirements for transformers, fuses, indicating lights, heat tracing, electric heaters, and recombiners with respect to the definitions, background, and specific electrical examples in the license renewal rule (circuit breakers, relays, motors, circuit boards, etc.). Based on the considerable discussion provided in the rule and SOC, the staff compared the electrical components identified above with the examples explicitly provided in the rule in terms of how the performance of their intended functions would be achieved and whether aging degradation of these components would be readily monitored using currently available techniques, in a similar way by which the examples in the rule (circuit breakers, relays, switches, etc.) would be monitored. These techniques include performance or condition monitoring by testing and maintenance/surveillance programs that include instrument checks, functional tests, calibration functional tests, and response time verification tests. The results of these tests and performance monitoring programs can be analyzed and trended to provide an Indication of aging degradation for these electrical components as discussed below:

* Transformers perform their intended function through a change in state by stepping down voltage from a higher to a lower value, stepping up voltage to a higher value, or providing isolation to a load. Transformers perform their intended function through a change in state similar to switchgear, power supplies, battery chargers, and power inverters, which have been excluded in §54.21(a)(1)(i) from an aging management review. Any degradation of the transformer's ability to perform its intended function is readily monitorable by a change in the electrical performance of the transformer and the associated circuits. Trending electrical parameters measured during transformer surveillance and maintenance such as Doble test results, and advanced monitoring methods such as infrared thermography, and electrical circuit characterization and diagnosis provide a direct indication of the performance of the transformer. Therefore, transformers are not subject to an aging management review.

* Indicating lights (dual filament) perform their intended function through a change in state by displaying readily monitorable visible light when energized with sufficient voltage. Indicating lights perform their intended function through a change in state similar to transistors and circuit boards, which have been excluded in §54.21(a)(1)(i) from an aging management review. Any degradation of the indicating lights ability to perform its intended function is readily monitorable since the lights (e.g., control room and local panel annunciators) typically have both a visual and audio test capability that is initiated on a periodic basis by the operator. This self-test capability is relied upon to provide a direct indication of the performance of the indicating lights. Therefore, indicating lights are not subject to an aging management review.

* Heat tracing performs its intended function through a change in state by supplying heat when energized, for example, to a boric acid system or a

refueling water storage tank/piping in order to maintain a minimum solution temperature to prevent boron from precipitating out or water from freezing in an outside pipe. Heat tracing performs its intended function through a change in state when energized similar to a power supply, battery charger, power inverter9 etc., which have been excluded in §54.21(a)(1)(i) from an aging management review. Any degradation of the heat tracing to perform its intended function is readily monitored by alarm circuitry (control room and local panel annunciators) or by surveillance requirements that monitor solution temperature on a periodic basis which provides a direct indication of the performance of the heat tracing. Therefore, heat tracing is not subject to an aging management review.

* Electric heaters perform their intended function through a change in state by supplying heat when energized, for example, to a pressurizer water volume for reactor coolant system pressure control. Electric heaters perform their intended function through a change in state similar to a battery charger, power inverter9 power supply, etc., that change state when energized and which have been excluded in §54.Z1(a)(1)(i) from an aging management review. Any degradation of the electric heaters' ability to perform their Intended function due to aging will be readily monitorable from existing monitoring equipment (voltmeters and active performance of the equipment in the circuit) and surveillance requirements by verifying that the heaters are energized and by measuring circuit current on a periodic basis. Therefore, electric heaters are not subject to an aging management review for the intended function of supplying heat. The pressure boundary intended function would still be subject to an aging management review.

The staff has also considered the aging management review requirements for fuses and hydrogen recombiners as discussed below:

* Fuses perform one of their two intended functions through a change in configuration dr state of the fuse by interrupting power In the case of a fault or overload in a load in order to provide protection to the rest of the electrical circuit. Fuses also perform a second intended function which is to maintain electrical continuity during non-faulted conditions. Unlike other electrical components which have simliar continuity functions such as breakers, switches, and relays which have been excluded in § 54.21 (a)(1)(i) from an aging management review, degradation of the fuse's ability to perform this intended function due aging Is not readily monitorable. Degradation of the fuse's intended continuity function may not result in detectable losses in associated system safety functions until degradation becomes unacceptable. Therefore, the staff believes that fuses are subject to an aging management review.

* Recombiners remove gaseous hydrogen from the containment atmosphere by combining hydrogen with oxygen to form water. This intended function is accomplished with several component types such as electric heater

banks, cabling, connections, etc. As such, recombiners should be considered as complex assemblies and should be evaluated on a plant-specific basis to determine If they are subject to an aging management review for renewal.

Based on the above assessment, the staff concluded that these components, with the exception of fuses and recombiners, perform their intended function(s) with a change in configuration/state and the effects of aging are readily monitored and therefore. are not subject to an aging management review. Electrical and I&C structures and components that are subject to an aging management review for renewal include, but may not be limited to: electrical cables and connections, fuses, electrical and I&C penetration assemblies, cable trays, and electrical and I&C cabinets, panels, racks, frames, enclosures, and other similar component supports.

NRC staff recommendations:

The NRC staff recommends revising Appendix B of NEI 95-10 to indicate that transformers, indicating lights, heat tracing, and electric heaters do not require an aging management review (recombiners should remain plant-specific) and to state that electrical and I&C structures and components subject to an aging management review *for* renewal should include: electrical cables and connections, fuses, electrical and I&C penetration assemblies, cable trays, and electrical and I&C cabinets, panels, racks, frames, enclosures, and other similar component supports.

REFERENCE 3

LICENSE RENEWAL ISSUE NO. 98-0016, "AGING MANAGEMENT REVIEW OF FUSES," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated April 27, 1999

April 27, 1999

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, N.W, Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0016, "AGING MANAGEMENT REVIEW OF FUSES"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution for the subject issue. The staff plans to implement the recommended resolution as part of the next revision to the " Standard Review Plan for License Renewal." We also expect NEI 95-10, "Industry Guideline for Implementating the Requirements of 10 CFR Part 54 - The License Renewal Rule," to be revised to reflect the guidance provided in that attached staff position. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Robert Prato at 301-415-1147.

Sincerely,

/Signed/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project: 690

Enclosure: As stated

cc w/enclosure: See next page

Background

On September 19, 1997, the staff issued a position (Attachment 1) on five electrical components including fuses. On April 10, 1998, NEI issued a response (Attachment 2) to this position. In this letter, NEI agreed with the staff's positions with the exception of the staff's determination that fuses require an aging management review. In response to NEI's position, the staff reviewed its determination that fuses are passive, long-lived components requiring an aging management review. The following are the results of the staff's review and its current position on the matter of fuses.

Evaluation

The April 10, 1998, letter, contains the following conclusions:

1. "Table 4.1-1in NEI 95-10 is entitled "Typical Passive Structure and Component Intended Functions." The purpose of this table is to identify typical intended functions for long-lived passive structures and components. As a result of discussions between NEI and the NRC staff we specifically included the intended function, "Provide insulation resistance to preclude shorts, grounds and unacceptable leakage current," to address electrical cables and connections. Electrical cables and connections are the only two items identified in §54.21(a)(1)(i) that require an aging management review because they are "passive" components. Electrical continuity is not included as an intended function of electrical cables and connections. Therefore, it is our view that "electrical continuity" is not an intended function of other components identified."

2. "It is our position that electrical continuity is not an intended function" of fuses.

The staff disagrees with the general conclusion that "electrical continuity" is not an intended function of electrical components as is stated in item 1. In its April 10, 1998 letter, NEI stated that *"continuity during non-fault conditions is a function of all electrical components."* The staff agrees that continuity is a function of most electrical components, including fuses, that should be assessed for its importance to license renewal prior to making a determination that an aging management review is not required. The following is the basis for the staff's conclusion:

- Table 4.1-1 is a list of typical intended functions and was never intended to be all inclusive.

- The list of structures and components requiring an aging management review under §54.21(a)(1)(i) is also not intended to be a complete list of "passive" structures and components. The rule clearly states that the list in question *"include, but are not limited to"* the structures and components contained in that list.

Enclosure

- Finally, the Statements of Consideration contains the following discussion:

  The previous license renewal rule required an applicant for license renewal to identify, from systems, structures, and components important to license renewal, those structures and components that contribute to the performance of a "required function" or could, if they fail, prevent systems, structures, and components from performing a "required function." This requirement initially posed some difficulty in conducting pre-application reviews of proposed scoping methodologies because it was not clear what was meant by "required function." Most systems, structures, and components have more than one function and each could be regarded as "required." Although the Commission could have required a licensee to ensure all functions of a system, structure, or component as part of the aging management review, the Commission concluded that this requirement would be unreasonable and inconsistent with the Commission's original intent to focus only on those systems, structures, and components of primary importance to safety. Consideration of ancillary functions would expand the scope of the license renewal review beyond the Commission's intent. Therefore, the Commission determined that "required function" in the previous license renewal rule refers to those functions that are responsible for causing the systems, structures, and components to be considered important to license renewal.

In the SOC, the Commission distinguished between functions that are of primary importance to safety and those that may be ancillary. Fuses may perform both kinds of functions. The staff has evaluated whether fuses require an aging management review, based on its applications in Nuclear Power Plant electrical systems and the two distinct functions they may perform.

1. A fuse can be included in an electrical system to provide a function directly related to nuclear power plant safety such as containment integrity protection (*i.e.* to limit fault damage to a containment electrical penetration) or to provide isolation protection for the Class 1E portion of the electrical system (*i.e.* to protect Class 1E electric equipment from faults originating in non-Class 1E equipment). Fuses included in nuclear power plant systems to perform such functions are intended to prevent or mitigate the consequences of accidents that could result in potential exposure comparable to the guidelines in § 50.34(a)(1) or § 100.11 of the Commission's regulations. Such fuses perform functions that are defined as "safety-related" in 10 CFR § 54.4(a)(1), and are, therefore, within the scope of license renewal.

Fuses having the intended safety-related functions identified above perform those functions with a change in configuration and, pursuant to 10 CFR 54.21(a)(1)(i), are not subject to an aging management review. The continuity function of such fuses, however, is not the reason for their inclusion in nuclear power plant systems. Rather, the isolation function of these fuses is of primary importance to safety and the reason for their inclusion in systems. Continuity is merely an ancillary function in these applications. Accordingly, such fuses do not require an aging management review.

It should be noted that the staff also considered potential aging mechanisms that may prevent a fuse from completing its safety-related fault protection function. Because of the fact that a change in configuration is required in the performance of this function, the staff will not pursue this concern under 10 CFR Part 54. However, because of its potential safety significance, and the fact that this concern may be equally important to current licensing terms, the staff intends to assess this issue to determine if it should be a Generic Safety Issue under 10 CFR Part 50.

(2) A fuse may also be included in an electrical system solely to limit the potential extent of fault damage (*e.g.* branch circuit protection) and thus increase the availability or reliability of the overall electrical system. Such fuses are installed essentially as equipment protection devices. Such fuses perform this function with a change in configuration as in (1) above and are not subject to an aging management review.

In addition, the continuity function of such fuses is not the reason for their inclusion in nuclear power plant systems. As such, the continuity function is merely an ancillary function in these applications. Accordingly, such fuses do not require an aging management review.

Conclusion

As set forth above, the staff has concluded that fuses do not require an aging management review under 10 CFR 54.21(a)(1). This item is considered resolved.

REFERENCE 4

LICENSE RENEWAL ISSUE NO. 98-0100, "CREDITING FERC-REQUIRED INSPECTION AND MAINTENANCE PROGRAMS FOR DAM AGING MANAGEMENT," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated May 5, 1999

May 5, 1999

Mr. Douglas J. Walter
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0100, "CREDITING FERC-REQUIRED INSPECTION AND MAINTENANCE PROGRAMS FOR DAM AGING MANAGEMENT"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution for the subject issue. The staff plans to implement the recommended resolution as part of the next revision to the draft Regulatory Guide entitled "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses." We also expect NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," to include the necessary changes to reflect the enclosed guidance. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you provide those comments to us in writing within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Samson Lee at 301-415-3109.

Sincerely,

/Signed/

Christopher I. Grimes, Chief
License Renewal & Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 690

Enclosure: As Stated

cc w/encl: See next page

LICENSE RENEWAL ISSUE NO. 98-0100
CREDITING FERC-REQUIRED INSPECTION
AND MAINTENANCE PROGRAMS FOR
DAM AGING MANAGEMENT

1. Introduction

The issue arose as to what type of program could be credited as a dam aging management program for the purposes of license renewal. Industry has asked whether simply citing an inspection program performed to meet Federal Energy Regulatory Commission (FERC) or other regulatory agency requirements would be adequate to demonstrate that dams will be maintained in accordance with the Current Licensing Basis (CLB) and therefore satisfy the requirements under Title 10 of the Code of Federal Regulations (10 CFR), Section 54.21.

2. Background

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As a part of the Integrated Plant Assessment performed for the license renewal application of the Oconee Nuclear Station, Duke Energy Corporation identified earthen embankments, dams, and related structures as being subject to Aging Management Review (AMR). They also identified a series of potential aging effects for those structures and claimed existing inspection programs, either the FERC required Five-Year Inspection or the Duke Power Five-Year Underwater Inspection of Hydroelectric Dams and Appurtenances, manage those effects. The application stated that a regular program of inspections, coupled with planned corrective actions, to be implemented should any deficiencies be discovered, should be adequate to safely maintain a dam and its appurtenances indefinitely.

Many dams on nuclear sites are already subject to periodic inspection due to the Federal Dam Safety Program which was initiated in 1977. This program, developed in response to several fatal dam failures in the 1970's, encourages strict safety standards in the practices and procedures employed by Federal agencies or by dam owners regulated by Federal agencies with regard to dam design, construction, inspection, maintenance, and management. The NRC relies on FERC to perform safety inspections of dams for which the NRC is responsible under this Federal dam safety program.

3. Discussion

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Dam Aging Management

The requirements for an application for license renewal for a nuclear power plant are specified in 10 CFR Part 54, specifically, Section 54.21(a)(3):

> *For each structure and component identified...[in the Integrated Plant Assessment in the application, the applicant must] demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.*

Enclosure

Since dam aging effects are related to material loss, damage, or movement due to erosion, corrosion, settlement, leakage, internal stresses, and other sources, a visual inspection of the external surfaces of a dam above and below water lines should detect any significant aging effects. Once detected, corrective actions can generally be taken to rectify the problem and minimize further degradation. Continued regular inspections coupled with a maintenance/corrective action program would be expected to keep a dam functioning safely during the period of extended operation associated with license renewal.

To that end, the continuation of a proper inspection program into the period of license extension should be adequate for dam aging management. What constitutes a proper inspection program and the crediting of programs under regulatory jurisdiction are discussed below.

Aging Management Programs For Dams Under FERC Oversight

In May of 1997, the NRC staff issued a Commission paper (SECY-97-110) discussing the status of development of its own Dam Safety Program Plan for dams that fall under NRC jurisdiction. Currently, only 19 of the dozens of dams and related structures associated with, or located near, nuclear power or uranium mine facilities are under NRC purview. In this paper, the NRC stated it had undertaken activities to fully implement a formal dam safety program plan in compliance with the Federal Guidelines on Dam Safety.

Under this program, independent reviews, at various stages in the life cycle of an NRC jurisdictional dam are required. As stated in the Plan:

> *By nature, the concept of the owner performing the major functions of, and addressing the elements of, a dam-safety program, with regulatory agency overview, will meet the goal of the Federal Guidelines. For existing dams, the Federal Guidelines prescribe formal inspections at intervals not to exceed five years. For this program, owners will have to have such reviews and inspections conducted by a team of qualified individuals, with a majority of the members being independent of the owner's organization.*

The Plan also says:

> *The inspection criteria, frequency, and scope of the inspections shall, as a minimum, meet the Federal Guidelines. The frequency and scope of the inspections will be the resultant of those inspections conducted by the dam owners, combined with those of NRC, as the regulatory agency and those conducted by a State, if conducted under an acceptable dam-safety program. Recognition of State dam-safety programs as the regulatory control will only be made after a formal Memorandum of Understanding (MOU) has been executed between a specific State and NRC.*

In SECY-97-110, the NRC staff describes an agreement established between the NRC and FERC that provides for FERC assistance in inspecting dams under NRC jurisdiction. The dam safety strategy set forth in SECY-97-110 applies only to those 19 dams and structures under the jurisdiction of the NRC, and not to the many dams associated with nuclear power plants under the purview of other agencies. However, this NRC dam policy does recognize the expertise of FERC in the dam safety, inspection, and maintenance field.

In addition, as stated before, inspections, coupled with a maintenance/corrective action program, are an acceptable manner of managing degradation of dams. Therefore, for earthen embankments, dams, and related structures identified as being subject to AMR, the staff concludes that continued compliance with the requirements of FERC into the license renewal period, by virtue of that agency's authority and responsibility for ensuring that its regulated projects are constructed, operated, and maintained to protect life, health, and property, will constitute an acceptable dam aging management program for the purposes of license renewal.

In order to credit the inspection programs performed under FERC oversight, and to provide the demonstration required by §54.21(a)(3), a license renewal applicant should idicate that its dam is under FERC jurisdiction and that its inspection and maintenance probram is in conformance with FERC requirements.

Aging Management Programs For Dams Under Other Regulatory Agencies

In addition to FERC, there are several possible government entities (Federal, state, local) that may have regulatory authority over dams and government entity-approved private firms that may perform inspections. SECY-97-110 and the Dam Safety Program Plan generally conclude that programs under the direct supervision of FERC are assumed to be acceptable while programs implemented by other agencies (including the utility itself, a state regulatory agency, etc.) must be demonstrated to meet particular requirements.

The Army Corps of Engineers, by virtue of its extensive experience in the field of dam construction, maintenance, inspection, and regulation, is also recognized as expert in the field of dam safety. Inspection and maintenance programs under the purview of the Army Corps of Engineers, continued into the period of license renewal, would constitute an acceptable dam aging management program. Therefore, a license renewal application can similarly credit an inspection program under the Army Corps of Engineers to satisfy the demonstration required by §54.21(a)(3), by stating that the Corps has jurisdiction over the dam, and that the applicant's program is in conformance with Corps requirements.

While dams, embankment, and appurtenance inspection and maintenance programs that fall under a regulatory agency other than FERC or the Corps, may be comparably acceptable, they are not as well recognized, understood and documented. Therefore, these programs need to be described in the application and evaluated like the general (non-regulatory) aging management programs described below.

Not all dams at nuclear power plants fall under the jurisdiction of a regulatory or independent entity. Many dam inspection and maintenance programs administered by licensees are modeled after Federal agency programs, but are completely controlled and administered by the licensee.

Programs that are not conducted under the direct supervision of FERC or the Army Corps of Engineers will be evaluated for the attributes of effective aging management in accordance with the guidelines developed for implementing the license renewal review.

Specifically, the staff will review these programs in accordance with §54.21(a)(3) to determine whether they contain the essential elements needed to provide adequate aging management for dams. The dam programs and procedures will be evaluated against the following elements: (1) scope of program; (2) preventive actions; (3) monitoring, detecting, and trending; (4) acceptance criteria; and (5) administrative controls. Applicants will be expected to provide an appropriate program description to address these attributes. Inspection and maintenance programs similar to those under the jurisdiction of FERC or the Army Corps of Engineers are likely to satisfy the elements.

4. Resolution

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It is the staff's opinion that dam inspection and maintenance programs under the jurisdiction of FERC or the Army Corps of Engineers, continued through the period of the license renewal, will be adequate for the purpose of aging management. For programs not falling under the regulatory jurisdiction of FERC or the Army Corps of Engineers, the staff will evaluate the effectiveness of the aging management program based on comparability to the common practices of the FERC and Corps programs.

In addition, the applicant must include a description of its dam inspection program in its Final Safety Analysis Report supplement pursuant to §54.21(d), if it does not already exist.

The staff recommends that NEI 95-10 be revised to reflect this guidance, and the staff will include comparable guidance in the appropriate draft Standard Review Plan section.

REFERENCE 5

LICENSE RENEWAL ISSUE NO. 98-0104, "ACCEPTANCE REVIEW OF LICENSE RENEWAL APPLICATIONS," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated February 1, 2000

February 1, 2000

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE No. 98-0104, "ACCEPTANCE REVIEW OF LICENSE RENEWAL APPLICATIONS"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution for the subject issue. The staff plans to incorporate the recommended resolution as part of the next revision to the *Standard Review Plan for License Renewal*. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Sam Lee at 301-415-3109 or Hai-Boh Wang at 301-415-2958.

Sincerely,

/RA/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690

Enclosure: As stated

cc w/encl: See next page

LICENSE RENEWAL ISSUE No. 98-0104
INCORPORATION OF LESSONS LEARNED INTO SECTION 1.1
OF THE STANDARD REVIEW PLAN FOR LICENSE RENEWAL

1. BACKGROUND

Section 1.1 of the draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) guides NRC reviewers in determining whether submitted renewal applications are acceptable for docketing and whether they are timely and sufficient.

2. EVALUATION

The NRC staff has gained experience on the use of SRP-LR, Section 1.1, and the associated checklist during the acceptance reviews of the Calvert Cliffs and Oconee license renewal applications. The NRC staff identified potential enhancements to the SRP-LR and have incorporated these lessons learned into a proposed revision to Section 1.1 of the SRP-LR. The proposed revision is attached.

3. RESOLUTION

The staff is proposing to revise Section 1.1 of the Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants to incorporate lessons learned during NRC staff acceptance reviews of the first two license renewal applications. The proposed revision is attached.

ATTACHMENTS: 1 - PROPOSED REVISION TO SRP-LR SECTION 1.1

1.1 Docketing of Timely and Sufficient Renewal Application

REVIEW RESPONSIBILITIES

PRIMARY-BRANCH RESPONSIBLE FOR LICENSE RENEWAL PROJECTS

SECONDARY-BRANCH RESPONSIBLE FOR ENVIRONMENTAL REVIEW
BRANCHES RESPONSIBLE FOR TECHNICAL REVIEW, AS APPROPRIATE

I. AREAS OF REVIEW

This review plan section addresses the review of the acceptability of a license renewal application for docketing in accordance with 10 CFR 2.101 and whether a license renewal application is timely and sufficient in order to allow the provisions of 10 CFR 2.109(b) to apply. 10 CFR 2.109(b) was written to comply with the Administrative Procedures Act. Allowing 10 CFR 2.109(b) to apply to the application means that the current license will not expire until the NRC makes a final determination on the license renewal application.

It is important to note that this review is not a detailed in-depth review of the technical aspects of the application. Docketing of a timely and sufficient renewal application does not preclude requesting additional information as the review proceeds; nor does it predict the NRC's final determination regarding the acceptance or rejection of the renewal application. It is also important to note that a plant's current license will not expire after the passing of the license's expiration date if a timely and sufficient renewal application has been docketed. During this time until the renewal application has been finally determined by the NRC, the licensee must continue to comply with its licensing basis, including all applicable license conditions, orders, and rules and regulations.

The following areas relating to the license renewal application are reviewed:

A. Docketing/Sufficiency of Application

The license renewal application is reviewed for acceptability for docketing as a sufficient application in accordance with 10 CFR 2.101 and 10 CFR 2.109(b).

B. Timeliness of Application

The timeliness of a license renewal application is reviewed for applicability of 10 CFR 2.109(b) and 54.17(c).

II. ACCEPTANCE CRITERIA

The acceptance criteria for the areas of review define methods for meeting the requirements of the Commission's regulations in 10 CFR 2.101 and 10 CFR 2.109(b).

A. Docketing/Sufficiency of Application

It is enough that the licensee submits the required reports, analysis, and other documents required in such application (56 FR 64923). The same acceptance criteria apply to the docketing acceptance review of 10 CFR 2.101(a)(2).

B. Timeliness of Application

A sufficient license renewal application is timely if it is submitted at least 5 years, but not more than 20 years, before the expiration of the current operating license.

III. REVIEW PROCEDURES

A licensee may choose to submit plant-specific reports addressing portions of the license renewal rule requirements for NRC review and approval prior to submitting a renewal application. An applicant may incorporate by reference these reports or other information contained in previous applications for licenses or license amendments, statements, or correspondence filed with the Commission, provided that the references are clear and specific. However, the final determination of the docketing of a timely and sufficient renewal application is made only after a formal renewal application has been tendered to the NRC.

For each area of review, the following review procedures are to be followed:

A. Docketing/Sufficiency of Application

Upon receipt of a tendered application for license renewal, the reviewer should determine whether the applicant has made a reasonable effort to provide the administrative, technical, and environmental information. Draft Regulatory Guide DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,"[(1)] was issued for public comment on August 26, 1996 (61 FR 43792). DG-1047 provides draft guidance on the format and content of a renewal application. The reviewer should use the review checklist in Table 1.1-1 of this review plan section to determine whether the application is reasonably complete and conforms to the requirements in 10 CFR Part 54.

Items I.1 through I.10 in the checklist address administrative information and, for the purpose of this docketing/sufficiency review, the reviewer should check the "Yes" column if the information is included in the application. Item II in the checklist addresses timeliness of the application.

Items III.1 through III.4 and Item IV in the checklist address technical information and technical specification changes. The reviewer may consult Sections 3.0 and 4.0 of this Standard Review Plan (SRP) for information regarding a technical review. Although the purpose of this docketing/sufficiency review is not to determine the technical adequacy of the application, the reviewer should determine whether the applicant has provided reasonably complete information in the application to address the renewal rule requirements. The reviewer may request assistance from appropriate technical review branches to determine whether the application is reasonable in addressing the items in the checklist such that there is sufficient information in the application for the staff to begin its technical review. The reviewer would check the "Yes" column for a checklist item if the applicant has provided reasonably complete information in the application to address the checklist item.

Item V in the checklist addresses environmental information. The environmental review staff should review the supplement to the environmental report in accordance with the guidelines in Draft Regulatory Guide DG-4005, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses,"[2] which is the draft environmental regulatory guide for 10 CFR Part 51. The reviewer would check the "Yes" column if the staff in RGEB determines that the renewal application contains information consistent with the guidelines in the draft environmental regulatory guide. The NRC staff review and the NRC staff preparation of the supplemental environmental impact statement will be guided by Supplement 1, "Operating License Renewal for Nuclear Plants," to NUREG-1555.[3]

The application should address each item in the checklist for it to be a reasonably complete and sufficient application. If the reviewer determines that an item in the checklist is not applicable, the reviewer should include a brief statement that the item is not applicable and provide the basis for the statement.

If information in the application for a checklist item is either not provided or not reasonably complete and no justification is provided, the reviewer would check the "No" column for that checklist item. By checking the "No" column for any checklist item in Table 1.1-1, except as discussed in Section III.B, the reviewer indicates that the application is not acceptable for docketing as a sufficient renewal application, unless the applicant modifies the application to provide the specific information.

If the staff determines that the application is not acceptable for docketing as a sufficient application, the staff's letter to the applicant should clearly state that the application is not sufficient and is not acceptable for docketing, and that the provisions in 10 CFR 2.109(b) are not satisfied and the current license will expire at its expiration date. Further, the staff should discuss the deficiencies found in the application and offer an opportunity for the applicant to modify its application to provide the specific information. The staff would review the modified application, when submitted, to determine whether it is acceptable for docketing as a sufficient application.

If the reviewer is able to answer "Yes" to the applicable items in the checklist, the application is acceptable for docketing as a timely and sufficient renewal application. Therefore, the provisions of 10 CFR 2.109(b) are satisfied and the current license will not expire until the NRC makes a final determination on the renewal application. The staff would issue a letter to the applicant documenting the staff's determination that the application is acceptable for docketing as a timely and sufficient renewal application. Normally, this letter should be issued within 30 days of receipt of a renewal application. A notice of acceptance for docketing of the application and notice of opportunity for a hearing regarding renewal of licenses would then be published in the Federal Register.

If the staff determines that the application is acceptable for docketing as a sufficient application, the staff would begin its technical review. For license renewal applications, the NRC intends to maintain the docket number of the operating license in effect to ensure continuation of the requirements in the current licensing basis (CLB).

B. Timeliness of Application

Upon receipt of a tendered application for license renewal, the reviewer performs a docketing/sufficiency review, as discussed in Subsection III.A of this reveiw plan section. If the reviewer determines that the application is acceptable for docketing as a sufficient application, the reviewer should determine whether this application is submitted in a timely manner to meet the provisions of 10 CFR 2.109(b).

If the sufficient application is submitted at least 5 years before the expiration of the current operating license, the reviewer would check the "Yes" column in Item II in the checklist in Table 1.1-1. If an applicant has to modify its application, as discussed in Subsection III.A of this review plan section, before the staff can find the application acceptable for docketing as a sufficient application, the modified application should be submitted at least 5 years before the expiration of the current operating license.

If the reviewer checks the "No" column in Item II in the checklist indicating that a sufficient renewal application is not submitted at least 5 years before the expiration of the current operating license, the staff's letter to the applicant should clearly state that the application is not timely and that the provisions in 10 CFR 2.109(b) are not satisfied and the current license will expire at its expiration date. However, if the application is otherwise determined to be acceptable for docketing, the staff technical review would continue.

IV. EVALUATION FINDINGS

The reviewer determines if sufficient and adequate information has been provided to satisfy the provisions of this SRP section. Depending on the results of this review, one of the following conclusions is included in the staff's letter to the applicant:

- The NRC staff has determined that the applicant has submitted sufficient information that is complete and acceptable for docketing, in accordance with 10 CFR 54.19, 54.21, 54.22, 54.23, AND 51.53(c). However, the staff's acceptance and sufficiency determination does not preclude request for additional information as the review proceeds.

- The application is not acceptable for docketing as a timely and sufficient renewal application.

V. IMPLEMENTATION

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. Draft Regulatory Guide DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," August 1996.

2. Draft Regulatory Guide DG-4005, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses," July 1998.

3 Draft Regulatory Guide DG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," July 1997.

TABLE 1.1-1.
ACCEPTANCE REVIEW CHECKLIST FOR DOCKETING OF
TIMELY AND SUFFICIENT RENEWAL APPLICATION

		Yes	No
I.	General Information		
1.	Application identifies specific unit(s) applying for license renewal	___	___
2.	Filing of renewal application [10 CFR 54.17(a)] is in accordance with:		
A.	10 CFR Part 2, Subpart A		
a.	10 CFR 2.101	___	___
b.	10 CFR 2.109(b)	___	___
B.	10 CFR 50.4		
a.	The application is addressed to the Document Control Desk as specified in 10 CFR 50.4(a)	___	___
b.	The signed original application and 13 copies are provided to the Document Control Desk. One copy provided to the appropriate Regional office [10 CFR 50.4(b)(3)]	___	___
c.	Verify that the form of the application meets the requirements of 10 CFR 50.4(c)	___	___
C.	10 CFR 50.30		
a.	Application filed in accordance with 10 CFR 50.4 [10 CFR 50.30(a)(1)]	___	___
b.	Application submitted under oath or affirmation [10 CFR 50.30(b)]	___	___

TABLE 1.1-1.
ACCEPTANCE REVIEW CHECKLIST FOR DOCKETING OF TIMELY AND SUFFICIENT RENEWAL APPLICATION (Continued)

		Yes	No
3.	Applicant is eligible to apply for a license, and is not a foreign-owned or foreign-controlled entity [10 CFR 54.17(b)]	___	___
4.	Application is not submitted earlier than 20 years before expiration of current license [10 CFR 54.17(c)]	___	___
5.	Renewal application states whether it contains applications for other kinds of licenses [10 CFR 54.17(d)]	___	___
6.	Information incorporated by reference in the application is contained in other documents previously filed with the Commission, and the references are clear and specific [10 CFR 54.17(e)]	___	___
7.	Restricted data agreement is present and complies with 10 CFR 50.33(j) [10 CFR 54.17(f)]	___	___
8.	Written agreement on the accessibility of restricted data is provided [10 CFR 54.17(g)]	___	___
9.	Information specified in 10 CFR 50.33(a) through (e), (h), and (i) is provided or referenced [10 CFR 54.19(a)]:		
	A. Name of applicant	___	___
	B. Address of applicant	___	___
	C. Business description	___	___
	D. Citizenship and ownership details	___	___
	E. License information	___	___
	F. Construction or alteration dates	___	___
	G. Regulatory agencies and local publications	___	___

TABLE 1.1-1.
ACCEPTANCE REVIEW CHECKLIST FOR DOCKETING OF TIMELY AND SUFFICIENT RENEWAL APPLICATION (Continued)

	Yes	No
10. Conforming changes have been submitted to the standard indemnity agreement (10 CFR 140.92, Appendix B) to account for the proposed change in the expiration date [10 CFR 54.19(b)]	___	___
II. Timeliness Provision		
Sufficient application is submitted greater than 5 years before expiration of current license [10 CFR 2.109(b). If not, application can be accepted for docketing but timely renewal provision in 10 CFR 2.109(b) does not apply	___	___
III. Technical Information		
1. An integrated plant assessment [10 CFR 54.21(a)] consists of:		
A. For those systems, structures, and components within the scope of license renewal [10 CFR 54.4], identification and listing of those structures and components that are subject to aging management review in accordance with 10 CFR 54.21(a)(1)(i) and (ii)		
a. Description of the boundary of the system or structure considered (if applicant initially scoped at the system or structure level). Within this boundary, identification of structures and components subject to aging management review. For commodity groups, description of basis for the grouping	___	___
b. Lists of structures, and components subject to an aging management review (AMR)	___	___
B. Description and justification of method used to identify structures and components subject to aging management review [10 CFR 54.21(a)(2)]	___	___

	Yes	No
C. Demonstration that the effects of aging will be adequately managed for each structure and component identified, so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation [10 CFR 54.21(a)(3)]		
a. Description of the structure and component intended function(s).	___	___
b. Identification of applicable aging effects based on materials, environment, operating experience, etc.	___	___
c. Aging management programs are identified and described	___	___
d. Demonstration of aging management provided	___	___
2. An evaluation of time-limited aging analyses (TLAAs) [10 CFR 54.21(c)] consists of:		
A. Listing of plant-specific TLAAs in accordance with the six criteria specified in 10 CFR 54.3 [10 CFR 54.21(c)(1)]	___	___
B. An evaluation of each identified TLAA using one of the three approaches specified in 10 CFR 54.21(c)(1)(i) to (iii)	___	___
3. All plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on a TLAA are listed, and evaluations justifying the continuation of these exemptions for the period of extended operation are provided [10 CFR 54.21(c)(2)]		

TABLE 1.1-1.
ACCEPTANCE REVIEW CHECKLIST FOR DOCKETING OF TIMELY AND SUFFICIENT RENEWAL APPLICATION (Continued)

		Yes	No
	A. Listing of plant-specific exemptions that are based on TLAAs as defined in 10 CFR 54.3 [10 CFR 54.21(c)(2)]	___	___
	B. An evaluation of each identified exemption justifying the continuation of these exemptions for the period of extended operation [10 CFR 54.21(c)(2)]	___	___
IV.	A final safety analysis report (FSAR) supplement [10 CFR 54.21(d)] contains the following information:		
	1. Summary description of the aging management programs and activities for managing the effects of aging	___	___
	2. Summary description of the evaluation of TLAAs	___	___
V.	Technical Specification Changes		
	Any technical specification changes necessary to manage the aging effects during the period of extended operation and their justifications are included in the application [10 CFR 54.22]	___	___
VI.	Environmental Information		
	Application includes a supplement to the environmental report that is in accordance with the requirements of Subpart A of 10 CFR Part 51 [10 CFR 54.23]	___	___

REFERENCE 6

GUIDANCE ON ADDRESING GSI-168 FOR LICENSE RENEWAL, Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated June 2, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Washington, D.C. 2055-0001

June 2, 1998

Mr. Doug Walters
Nuclear Energy Institute
1776 I Street, N.W.
Washington, DC 20006

SUBJECT: GUIDANCE ON ADDRESSING GSI-168 FOR LICENSE RENEWAL

Dear Mr. Walters:

Unresolved generic safety issues (GSIs) within the scope of aging management review or time-limited aging analyses are to be addressed in a license renewal application as stated by the Commission when the amended license renewal rule, 10 CFR Part 54, was issued (60 FR 22484). Recent Nuclear Regulatory Commission (NRC) staff guidance on evaluating GSIs was provided in a letter to NEI dated January 29, 1998. One GSI meeting the criteria for evaluation for license renewal is GSI-168, "Environmental Qualification of Electrical Equipment."

For license renewal, the Statements of Consideration (SOC) for the amended license renewal rule (60 FR 22484) state that resolution of a GSI generically is not necessary for the issuance of a renewed license. However, designation of an issue as a GSI does not exclude the issue from the scope of the aging management review or time-limited aging evaluation. The Commission went on to provide four approaches that could be used to satisfy the finding required by 10 CFR 54.29. These approaches have been incorporated into the industry's guidance document NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision O, that the staff proposed to endorse in its Draft Regulatory Guide, DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses."

With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time. A renewal applicant should monitor updates to NUREG-0933, "A Prioritization of Generic Safety Issues," for revisions to GSI-168 during the review of its application and supplement its license renewal application if the issues associated with GSI-168 become defined such that providing the options or pursuing one of the other approaches described in the SOC becomes feasible. Guidance on supplementing a license renewal application after submitted to address GSIs is provided in the January 29, 1998, GSI letter.

The guidance in this letter is provided for addressing GSI-168 in a license renewal application and is not intended to be applied in any other context. Additionally, this letter does not modify the requirements for a license renewal applicant to address EQ as a time-limited aging analysis in accordance with the requirements of 10 CFR 54.21(c) or to ensure continued compliance with 10 CFR 50.49 for the period of extended operation.

If there are any questions, please contact Steve Hoffman at 301-415-3245.

Sincerely,

Christopher I. Grimes, Director
License Renewal Project Directorate
Division of Reactor Program
Management
Office of Nuclear Reactor Regulation

Project No.: 690

cc: See next page
R. Gill, Duke
C. Pierce, SNC

REFERENCE 7

LICENSE RENEWAL ISSUE NO. 98-0051, "EVALUATION OF JURISDICTION OF ASME SECTION XI, SUBSECTIONS IWE AND IWF, FOR LICENSE RENEWAL," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated March 6, 2000

March 6, 2000

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0051, "EVALUATION OF JURISDICTION OF ASME SECTION XI, SUBSECTIONS IWE AND IWF, FOR LICENSE RENEWAL"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution of the subject issue. The staff plans to incorporate the recommended changes to the Standard Review Plan for License Renewal in a future revision. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Peter Kang at 301-415-2779, or Sam Lee at 301-415-3109.

Sincerely,

/RA/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690

Enclosures: As stated

cc w/encl: See next page

Mr. Douglas J. Walters

LICENSE RENEWAL ISSUE NO. 98-0051
EVALUATION OF JURISDICTION OF ASME SECTION XI, SUBSECTIONS IWE AND IWF, FOR LICENSE RENEWAL

1. BACKGROUND

On May 8, 1998, Nuclear Energy Institute (NEI) provided the staff with comments on the working draft of the "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated September 1997. For managing the aging effects of boiling-water reactor (BWR) containment structure components in Chapter 3.4, the draft standard review plan for license renewal (SRP-LR) states that ASME Section XI, Subsections IWE and IWF be used for lockup caused by mechanical wear while only Subsection IWE is used for loss of material caused by corrosion. NEI commented that ASME Section XI, Subsection IWF should be added in the Sections 3.4.II.C.10, "Acceptance Criteria" and 3.4.III.C.10 "Review Procedures" of the aforementioned Chapter 3.4 of the SRP-LR as an adequate program for managing the aging effects of loss of material.

The staff's rationale for managing the aging effects of the lockup and loss of material for BWR containment structures in the draft SRP-LR was based on the recommendations provided by NUREG-1611, "Aging Management of Nuclear Power Plant Containments for License Renewal," in which the lockup (page 46) of containment pressure retaining components and their supports is managed with the implementation of ASME Section XI, Subsections IWE and IWF, respectively while the loss of material (pages 33 and 34) of structural steel and liner is managed by Subsection IWE but it does not address any component supports.

2. EVALUATION

The scope of ASME Section XI, Subsection IWE provides the rules and requirements for inservice inspection, repair, and replacement of Class MC pressure retaining components and their integral attachments, and of metal shell and penetration liners of Class CC pressure retaining components and their integral attachments. ASME Section XI, Subsection IWF provides requirements for periodic examinations (inservice inspection of Class 1, 2, and 3) of metal support components, including downcomer bracings, column and saddle supports, seismic restraints, and vent system supports.

The staff has reviewed BWR containment structure components listed in Sections 3.4.II.C.10 and 3.4.III.C.10 of the draft SRP-LR for managing the aging effects of loss of material and finds that the list includes both pressure retaining components and support components such as vent system supports for Mark 1 steel containments. Therefore, the staff agrees with NEI that ASME Section XI, Subsection IWF should be added to the Section 3.4.II.C.10, "Acceptance Criteria" and 3.4.III.C.10, "Review Procedures," as acceptable aging management of component supports for "Loss of Material" in the draft SRP-LR.

3. RESOLUTION

The staff will revise the affected pages of Section 3.4.II.C.10 ("Acceptance Criteria") and Section 3.4.III.C.10 ("Review Procedures") of the draft SRP-LR to add ASME Section XI, Subsection IWF. On this basis, the staff considers License Renewal Issue No. 98-0051 (MA2399) on the jurisdiction of ASME Section XI, Subsections IWE and IWF, resolved.

REFERENCE 8

LICENSE RENEWAL ISSUE NO. 98-12, "CONSUMABLES," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated March 10, 2000

March 10, 2000

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-12, "CONSUMABLES"

Dear Mr. Walters:

Enclosure 1 is the staff's proposed resolution of the subject issue. Based on the December 8, 1999, meeting, as documented in the January 21, 2000, meeting summary (Enclosure 2), the staff concluded that the enclosed changes should be made to the draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants." It is expected that comparable changes will be made to NEI 95-10, "Industry Guidance for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule." If there are any industry comments on the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Stephen Koenick at 301-415-1239.

Sincerely,

/RA/

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project 690

Enclosures: As stated

cc w/encl: See next page

Proposed Changes to the SRP

Based on the discussion provided in the "Summary of December 8, 1999, Meeting on License Renewal Issue (LR) 98-12, 'Consumables'," dated January 21, 2000, the following should be added to Table 2.2-2, "Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment," of the draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants."

ITEM	CATEGORY	STRUCTURE, COMPONENT, OR COMMODITY GROUP	STRUCTURE, COMPONENT, OR COMMODITY GROUP MEETS 10 CFR 54.21(a)(1)(i) (YES/NO)
	Subcomponent	Packing, Gaskets, Components Seals, and O-rings	Yes[1]
	Subcomponent	Structural Sealants	Yes[2]
	Consumable	Oil, Grease, and Component Filters	No[3]
	Consumable	System Filters, Fire Extinguishers, Fire Hoses, and Air Packs	Yes[4]

1 These subcomponents would not necessarily be called out explicitly in the scoping and screening procedures. Instead they would be implicitly addressed at the component level. The applicant will be able to exclude these subcomponents utilizing a clear basis such as the example of ASME Section III not being relied upon for pressure boundary.

2 These subcomponents would not necessarily be called out explicitly in the scoping and screening procedures. Instead they would be implicitly addressed at the component level. Structural sealants may perform functions without moving parts or change in configuration and they are not typically replaced. It is expected that the applicant's structural aging management program will address these items with respect to an aging management review program on a plant-specific basis.

3 For these commodities, the screening process would be expected to exclude these materials because they are short-lived and are periodically replaced.

4 These components may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii) in that they are replaced on condition. The application should identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Enclosure 1

January 21, 2000

ORGANIZATION Nuclear Energy Institute (NEI)

SUBJECT: SUMMARY OF DECEMBER 8, 1999, MEETING ON LICENSE RENEWAL ISSUE (LR) 98-12, "CONSUMABLES"

The Nuclear Regulatory Commission (NRC) met with the NEI on December 8, 1999, to discuss LR 98-12, "Consumables." The agenda for the meeting is provided in Attachment 1. Attendees are listed in Attachment 2.

Background

As part of an effort to more efficiently resolve generic issues involved with license renewal, the NRC staff is in the process of implementing a informal process for resolving generic issues. This process will be outlined in NRR Office Letter No. 805, "License Renewal Application Review Process." To resolve the generic issues in which there is disagreement between stakeholders and NRC, the NRC is implementing an appeals process in which stakeholders and NRC staff have successive management meetings in order to identify resolution paths for the issues. The meeting on December 8, 1999, was a trial appeals meeting. The NRC issued a staff position on "consumables" in a letter dated April 21, 1999. In a letter from D. Walters of NEI to C. Grimes of NRC dated July 30, 1999, NEI articulated several disagreements with the NRC staff position.

Discussion

The meeting provided useful dialogue with consensus being reached in numerous areas. The result of the meeting is captured in this meeting summary. The outcome of this process is for the NRC staff to develop proposed guidance that will be incorporated into the working draft, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP). It would be expected that NEI would revise their industry document NEI 95-10, "Industry Guidance for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," accordingly. In order to ensure proper characterization of the dialogue between the stakeholders and staff, this meeting summary will be followed with a letter to interested stakeholder containing proposed guidance soliciting feedback. The meeting summary according to the agenda of the topics discussed at the meeting is as follows:

Enclosure 2

- The need to categorize consumables as components, piece parts, or subcomponents.

LR 98-12, "Consumables," represented an attempt to categorize various components, subcomponents, piece parts and other materials that are typically replaced during routine maintenance and testing, or based on component performance. The NRC staff position divided the "consumables" into four categories. There was agreement that the four categories represented different types of material that need to be addressed differently for the purpose of license renewal. Category A, comprises packing, gaskets, component seals, and o-rings, represent subcomponents and specific guidance is addressed below in item 3.a. Category B, comprises structural sealants, represent subcomponents that are treated differently from Category A in that they are long-lived components and may serve a passive function. Specific guidance is addressed in item 3.b. Category C, comprises oil, grease, and component filters, represent consumables that are short-lived. Specific guidance is provided in item 3.c. Category D, comprises of system filters, fire extinguishers, fire hoses, and air packs, represent components that are routinely replaced on condition. Specific guidance is provided in item 3.d.

With respect to the need to categorize consumables there was a general consensus to not exclusively categorize consumables as components, piece parts, or subcomponents. However, from a process consideration the following was discussed.

The "consumable" would not be explicitly called out in the scoping and screening procedures. Instead it would be implicitly included at the component level, (i.e., if a valve is identified as being in scope, a seal would be in scope as a subcomponent of the valve). The consumable will be considered during the aging management review. The methodology for performing the aging management review of the various subcomponents is a procedure that is maintained onsite and is auditable. It is in this procedure, in which the applicant can provide justification for excluding the specific "consumable" from scope.

- Reliance on performance or condition monitoring for generic exclusion.

There was mutual agreement between NRC and NEI that performance or condition monitoring cannot be used for generic exclusions, but this does not prevent it from being used for a site-specific justification.

- Component Replacement Strategy or Aging Management Program

    - Packing, Gaskets, Components Seals, and O-rings

For the purpose of addressing packing, gaskets, components seals, and o-rings during the review of a license renewal application, the reviewer should consider these items as subcomponents. These subcomponents would not be explicitly called out in the scoping and screening procedures. Instead they would be implicitly included at the component level, (i.e., if a valve is identified as being in scope, a seal would be in scope as a subcomponent of the valve). They will be considered during the aging management review. The methodology for performing the aging management review of the various subcomponents is a procedure that is maintained onsite and is auditable. For this category of "consumables" consistent with the staff position, the applicant will be able to exclude these components utilizing a clear basis such as the example identified in the NRC staff position of ASME Section III not being relied upon for pressure boundary.

This process of addressing this category of consumables during the aging management review should be summarized in the application during the methodology for conducting the aging management review.

- Structural sealants

For the purpose of addressing structural sealants during the review of a license renewal application, the reviewer should consider these items as subcomponents. These subcomponents would not be explicitly called out in the scoping and screening procedures. Instead they would be implicitly included at the component level. They will be considered during the aging management review. The methodology for performing the aging management review of the various subcomponents is a procedure that is maintained onsite and is auditable. For this category of "consumables" consistent with the staff position, structural sealants may perform functions without moving parts or change in configuration and they are not typically replaced on condition. It is expected that the applicant's structural aging management program will address these items with respect to an aging management review program on a plant specific basis.

This process of addressing this category of consumables during the aging management review should be summarized in the application during the methodology for conducting the aging management review.

- Oil, Grease, and Component Filters

For the purpose of addressing oil, grease, and component filters during the review of a license renewal application, the reviewer should consider these other materials as consumables that are short-lived. For this category of "consumables" consistent with the staff position, this material can be excluded on the basis of being short-lived and periodically replaced.

This process of addressing this category of consumables during the aging management review should be summarized in the application during the methodology for conducting the aging management review.

- System Filters, Fire Extinguishers, Fire Hoses, and Air Packs

For the purpose of addressing system filters, fire extinguishers, fire hoses, and air packs during the review of a license renewal application, the reviewer should consider these items as components. For this category of "consumables" consistent with the staff position, these components may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii) in that they are replaced on condition.

This process of addressing this category of consumables during the aging management review should be summarized in the application during the methodology for conducting the aging management review.

Conclusion

The proposed staff guidance will be developed based on the discussion above. This guidance will be incorporated into the SRP as it is revised. In accordance with the appeals process being developed, the interested stakeholders will have the opportunity to provide feedback to the proposed guidance. If notified, the specific disagreement with accompanying basis would be elevated to the next level of management. Without comment, the proposed guidance based on this meeting summary will represent resolution and closure of LR 98-12, "Consumables."

Stephen S. Koenick, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 690

Attachments: As stated

cc w/atts: See next page

Agenda for December 8, 1999, meeting on License Renewal Issue No. 98-12 (LR 98-12), "Consumables"

1. The need to categorize consumables as components, piece parts, or subcomponents.

2. Reliance on performance or condition monitoring for generic exclusion.

   U.S. Nuclear Regulatory Commission, "Nuclear Power Plant License Renewal; Revisions," *Federal Register*, Vol. 60, No. 88, Monday May 8, 1995, page 22478.

   It is important to note, however, that the Commission has decided not to generically exclude passive structures and components that are replaced based on performance or condition from an aging management review. Absent the specific nature of the performance or condition replacement criteria and the fact that the Commission has determined that components with "passive" functions are not as readily monitorable as components with active functions, such generic exclusions is not appropriate. However, the Commission does not intend to preclude a license renewal applicant from providing site-specific justification in a license renewal application that a replacement program on the basis of performance or condition for a passive structure or component provides reasonable assurance that the intended function of the passive structure or component will be maintained in the period of extended operation.

3. Component Replacement Strategy or Aging Management Program

   1. Packing, Gaskets, Components Seals, and O-rings

   2. Structural sealants

   3. Oil, Grease, and Component Filters

   4. System Filters, Fire Extinguishers, Fire Hoses, and Air Packs

Attachment 1

ATTENDANCE LIST
MEETING ON LICENSE RENEWAL ISSUE NO. 98-12 (LR 98-12), "CONSUMABLES"
DECEMBER 8, 1999

NAME	ORGANIZATION
BOB PRATO	NRC/NRR/DRIP/RLSB
JANICE MOORE	NRC/OGC
P.T. KUO	NRC/NRR/DRIP/RLSB
CHRIS GRIMES	NRC/NRR/DRIP/RLSB
GOUTAM BAGCHI	NRR/DE
BILL CORBIN	VIRGINIA POWER
FRED POLASKI	PECO-ENERGY
BERNIE VAN SANT	OMAHA PUBLIC POWER
JOHN RYCYNA	CONSTELLATION NUCLEAR SERVICES
DOUG WALTERS	NEI
STEVE HALE	FPL
STEPHEN KOENICK	NRC/NRR/DRIP.RLSB
JAKE ZIMMERMAN	NRC/NRR/DRIP/RLSB
STEVE HOFFMAN	NRC/NRR/DRIP/RLSB
LYNN CONNOR	DSA
KAMAL MANOLY	NRC/NRR/DE/EMEB
HANS ASHAR	NRC/NRR/DE/EMEB
WILLIAM BURTON	NRC/NRR/DRIP/RLSB
MELVIN FRANK	SCIENTECH/NUSIS
MICHAEL SEMMLER	DUKE ENERGY
HAI-BAH WANG	NRC/NRR/DRIP/RLSB
NANCY CHAPMAN	SERCH/BECHTEL
JIT VORA	NRC/RES/DET/MEB

Attachment 2

REFERENCE 9

LICENSE RENEWAL ISSUE NO. 98-0013, "DEGRADATION INDUCED HUMAN ACTIVIES," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated June 5, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 5, 1998

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 1 Street, N.W, Suite 300
Washington, D.C. 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0013, "DEGRADATION INDUCED HUMAN ACTIVITIES"

Dear Mr. Walters:

Attached is the staff's evaluation and proposed resolution for the subject issue. The staff plans to incorporate the recommended change to the Standard Review Plan for License Renewal in a future revision. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Robert Prato at 301-415-1147.

Sincerely,

Christopher I. Grimes, Director
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As Stated

cc: GLainas, NRR
RSpessard, NRR
GHolahan, NRR
JRoe, NRR
Licensee Renewal Steering Committee

LICENSE RENEWAL ISSUE NO. 13
DEGRADATION INDUCED BY HUMAN ACTIVITIES

BACKGROUND

By letter dated May 22, 1996, the Baltimore Gas and Electric Company (BGE) submitted five (5) Integrated Plant Assessment (IPA) System and Commodity Reports to the NRC staff for review and approval in accordance with Part 54, the license renewal rule. In one of these reports, "Appendix A, Technical Information, 7.6 Component Supports Commodity Evaluation," the staff found that Table 7.6-3, "Potential and Plausible ARDMs for Equipment Supports," of the report indicates that "Abuse, Impacts, Accidents" associated with human activities was designated as a potential age-related degradation mechanism (ARDM) that may need to be managed for license renewal. Further, in Section 2.2.9 of BGE report, "Aging Management Review Report for Component Supports," Revision 2, it states, *Literature and industry experience provide examples of component support degradation by abuse, impacts, or accidents. These events potentially cause immediate damage in which case they are not considered ARDMS. However, these events may also initiate gradual degradation in which case the initiating event is an ARDM. This gradual degradation is defined as 'error-induced aging degradation' by the NRC-approved Nuclear Power Plant Aging Terminology (Reference 2.7). The root cause of failures from error induced aging degradation is human error, not aging. However, the control of error induced aging degradation is part of aging management.*

Reference 2.7 in the forgoing quote refers to EPRI report TR-100844, "Nuclear Power Plant Common Aging Terminology," dated November 1992. While the NRC staff encouraged EPRI to publish that report as a way to improve communication on aging issues, the NRC did not approve the report as a means of implementing 1 0 CFR 54, which might be implied from the BGE conclusion.

This issue was addressed in a request for additional information (RAI) on August 30, 1996. By letter dated February 14, 1997, BGE submitted its response to the staff's RAI and deleted the ARDM "Abuse, Impacts, Accidents" (which is related to human activities) from the component support report. Additionally, by letter dated October 22, 1997, BGE submitted a revised response to the staffs RAI and removed the potential ARDM for "Abuse, Impacts, Accidents" associated with human activity from Table 3.1-3 (previously Table 7.6-3), "Potential and Plausible ARDMs for Component Supports."

Because of uncertainties in the role of Abuse, Impacts, Accidents in the completeness of the definition of aging effects under 10 CFR 54, the staff initiated a license renewal issue entitled: "Degradation Induced by Human Activities."

2. EVALUATION

The statement of considerations (SOC) for the license renewal rule, 1 0 CFR Part 54, dated May 8, 1995, states, *The Commission believes that, regardless of the specific aging mechanism, only age-related degradation that leads to degraded performance or condition (i.e. detrimental effects) during the period of extended operation is of principal concern for license renewal* (60 FR 22469). The SOC further indicates that

(1) aging is a continuous process, and (2) passive structures and components are those that perform an intended function without moving parts or without a change in configuration or properties (60 FR 22465 and 60 FR 22477, respectively).

The staff believes that degradation induced by human activity, including abuses, accidents, and specific or unexpected events, is not an aging effect that needs to be subject to an aging management review. If a safety-significant system, structure or component is accidentally damaged due to human activities, the staff would expect the licensee to take immediate corrective actions under 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. For example, the repeated bending of an electrical wire at a connection to disconnect and re-connect for maintenance activities should be addressed by routine maintenance practice including inspection of the wire for physical damage due to installation, operation, or maintenance.

Similarly, human activities might influence degradation or the rate of degradation by increasing air and water borne contaminants. However, this influence would be considered through the evaluation of the specific aging effects, such as chemical attack, and, therefore, do not need to be addressed as a separate aging effect.

Subsection III.B of Section 3.0 of the working draft Standard Review Plan for License Renewal (SRP-LR), dated September 1997, addresses aging effects from abnormal events as follows: *Aging effects from abnormal events need not be postulated specifically for renewal.* The SRP-LR discusses examples of abnormal events and states: For *example, abuse due to human error is an abnormal event and aging effects from such abuse need not be postulated for renewal.* The SRP-LR also provides an example of abnormal events whose contribution to the aging effects should be evaluated for license renewal by stating: *For example, if a resin intrusion has occurred in the reactor coolant system at the applicant's plant, the applicant should consider the contribution of this resin intrusion event to the aging effects, such as cracking, of the reactor coolant system components for renewal.*

3. RESOLUTION

Based on the above evaluation, the staff concludes that the issue of degradation induced by human activities need not be considered as a separate aging effect and should be excluded from an aging management review. The guidance in the working draft SRP-LR remains valid. However, the term "human error' should be replaced with "human activity" on Page 3.0-9 of the working draft SRP-LR. This issue is considered resolved.

REFERENCE 10

LICENSE RENEWAL ISSUE NO. 98-0014, "STAFF GUIDANCE FOR LICENSE RENEWAL APPLICATION SUBMITTALS ON TIME-LIMITED AGING ANALYSES FOR ENVIRONMENTAL QUALIFICATION," Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated September 24, 1998

Project No. 690

UNITED STATES
REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 1998

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE No. 98-0014, "STAFF GUIDANCE FOR LICENSE RENEWAL APPLICATION SUBMITTALS ON TIME-LIMITED AGING ANALYSES FOR ENVIRONMENTAL QUALIFICATION"

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution for the subject issue. The staff plans to implement the recommended resolution as part of the next revision to the draft Regulatory Guide entitled "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses." We also expect NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," to include the necessary changes to reflect the guidance provided in the enclosed guidance. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you provide those comments to us in writing within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. If you have any questions regarding this matter, please contact Robert Prato at 301-415-1147.

Sincerely,

Christopher I. Grimes, Project Director
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure: As stated

3

GUIDANCE ON ENVIRONMENTAL QUALIFICATION

TIME-LIMITED AGING ANALYSES

FOR LICENSE RENEWAL

An application for license renewal under 10 CFR Part 54 must contain an evaluation of time-limited aging analyses (TLAAs). TLAAs are defined in § 54.3 as *'those licensee calculations and analyses that (1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a); (2) Consider the effects of aging; (3) Involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) Were determined to be relevant by the licensee in making a safety determination; (5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and (6) Are contained or incorporated by reference in the CLB."* Most environmental qualification (EQ) analyses meet this definition.

In accordance with § 54.21 (c)(1), *"[a] list of time-limited aging analyses, as defined in § 54.3, must be provided"*. The TLAA list for EQ should include the components on the EQ Master List maintained pursuant to § 50.49(d) that are qualified for the full term (or longer) of the current operating license. The applicable option, as described below, should be identified for each item on the list.

For each TLAA, § 54.21 (c)(1) also requires that each applicant demonstrate one of following three options:

(c)(1)(i) *"The analyses remain valid for the period of extended operation."*

The applicant should summarize these analyses in the application for each item or group of electrical equipment. For the purposes of demonstration, these summaries should include (1) a summary of thermal and radiation analyses that show the existing analyses remain valid to the end of the period of extended operation and (2) the qualification calculation reference number. A thermal analysis summary should show that the bounding average ambient temperature used in the existing analysis is less than the calculated average ambient temperature based on preconditioning of the equipment for a 60-year qualified life. Each summary should also include the numerical values of these temperatures. A radiation analysis summary should show that the bounding 60-year integrated dose (normal dose plus design basis accident dose) is less than the dose used in the qualification of the equipment. Each summary should include the numerical values of these doses. Alternatively, the applicant may briefly explain why the equipment for which analyses remain valid is insensitive to aging.

(c)(1)(ii) *"The analyses have been projected to the end of the period of extended operation."*

The applicant should summarize these analyses in the application for each item or group of electrical equipment. For the purposes of demonstration, these summaries should include (1) a summary of thermal and radiation analyses that show the analysis was projected to the end of the period of extended operation and (2) the qualification calculation reference numbers. A thermal analysis summary should show that the bounding average ambient temperature, from actual temperature measurements, including self-heating, if appropriate, is less than the calculated average ambient temperature based on preconditioning of the equipment for a 60-year qualified life. Each summary should also include the numerical values of these temperatures. A radiation analysis summary should show that the bounding 60-year integrated dose (normal dose plus design basis accident dose) is less than the dose used in the original qualification of the equipment. Each summary should include the numerical values of these doses.

(c)(1)(iii) *"The effects of aging on the intended function(s) of EQ equipment will be adequately managed for the period of extended operation."*

Aging for the purposes of EQ, as identified under 10 CFR 50.49(e)(5), allows for replacement or refurbishment of all EQ components at the end of qualified life unless ongoing qualification demonstrates that the item has additional life. Therefore, for the purposes of license renewal, an aging management program for those components that are not qualified to the end of the period of extended operation should be placed in an ongoing qualification program, or replaced or refurbished at the end of its qualified life. For the purpose of demonstrating that the effects of aging will be managed, the application should include a description of activities for each item or group of electrical equipment, as well as any changes to the current licensing basis and plant modifications that are relied on to demonstrate that the intended function(s) will be adequately maintained despite the effects of aging during the period of extended operation.

Guidance for providing a description of the activities for an aging management program (i.e., refurbish or replacement) can be found in NEI 95-10, § 4.2.1.2, entitled 'Identify Plant Aging Management Programs.' In addition, in the Statements of Consideration accompanying the 1995 amendment to Part 54 (60 FR 22475), the Commission explained that "*the straightforward approach to detecting and mitigating the effects of aging begins with a process that verifies that the intended design functions of systems, structures, and components have not been compromised or degraded.*" In addition, guidance for the level-of-detail that should be provided in the application can be found in NEI 95-10, § 4.21.3, entitled "Demonstrating That the Effects of Aging are Managed," and § 6.2, entitled "Exhibit A - Technical Information." Toward that end, the demonstration should decide the process by which compliance with § 50.49 will be maintained throughout the period of extended operation. Particularly useful information regarding the process are those features related to the aging considerations described in § 50.49(e)(5), and decision criteria for replacing or refurbishing EQ equipment before the end of the equipment's qualified life.

If an applicant chooses to rely on testing as discussed under § 50.49(f) by applying ongoing test programs, the application should describe the program consistent with the guidance in IEEE 323-1974, Section 6.6 (1) or (2). In addition, the description should include the current qualified life for each item or groups of electrical equipment, and the period of time prior to the end of qualified life when testing will be completed.

In addition to the options discussed under 10 CFR 50.49, an applicant may want to re-analyze the qualified life of an item or group of electrical equipment at some time after the renewed license has been approved. For the staff to evaluate this option at the time of application, the applicant should define its aging management program by describing the attributes of the program that will be used in the re-analyses in the license renewal application. The attributes of re-analyses that will be performed should include analytical methods; data collection and reduction methods, if needed; underlying assumptions; acceptance criteria; corrective actions if acceptance criteria are not met; and the period of time prior to the end of qualified life when re-analyses will be completed. This information can be provided at a level-of-detail consistent with the guidance found in NEI 95-10, § 4.2.1.3, entitled "Demonstrating That the Effects of Aging are Managed," and § 6.2, entitled "Exhibit A - Technical Information."

The foregoing guidance describes an acceptable approach for satisfying the requirements of § 54.21 (c)(1) for EQ analyses. Applicants may propose alternate approaches, but such approaches should provide a comparable level of information so that the staff can conclude that the time-limited aging analyses for EQ have been identified and there is reasonable assurance that the effects of aging applicable to the EQ analyses will be adequately managed for the period of extended operation.

REFERENCE 11

GENERIC SAFETY ISSUES RELATED TO LICENSE RENEWAL (TAC NO. M92972), Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated January 29, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 29, 1998

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708

SUBJECT: GENERIC SAFETY ISSUES RELATED TO LICENSE RENEWAL (TAC NO. M92972)

Dear Mr. Walters:

In a letter dated October 21, 1996, the Nuclear Energy Institute (NEI) provided the results of an industry review of unresolved safety issues (USIs) and generic safety issues (GSIs) (collectively referred to herein as GSIs) applicable to license renewal. The NRC tracks GSIs in NUREG-0933, "A Prioritization of Generic Safety Issues," which is updated periodically. The NRC staff reviewed (1) the criteria NEI used for performing the review and (2) the list of GSIs identified by NEI as applicable to license renewal. On the basis of this review and further evaluation of the NUREG-0933 generic issues process, the staff has determined that the appropriate criteria for reviewing GSIs differs from those proposed by NEI. This letter responds to your October 21 letter and transmits the staff's guidance for performing the GSI review.

The staff has determined that all issues listed in NUREG-0933, Appendix B, with the following Safety Priority/Status classifications should be reviewed to identify any generic concerns that may be related to the effects of aging or time-limited aging analyses for systems, structures, or components within the scope of the license renewal rule:

- USI
- High priority
- Medium priority
- Note 1 "Possible Resolution Identified for Evaluation"
- Note 2 "Resolution Available (Documented in NUREG, NRC Memorandum, SER [safety evaluation report], or equivalent)"

These criteria are consistent, with one exception, with the staff's guidance in draft Branch Technical Position PDLR 3.0-1 (BTP), which is found in the September 1997 working draft of the standard review plan for license renewal. The evaluation of GSIs classified as Note 4, "Issue to be Prioritized in the Future." has been eliminated. Until the staff completes its prioritization process for Note 4 issues and documents the results in NUREG-0933, the full scope of Note 4 issues and the determination of whether safety concerns exist that warrant further pursuit (i.e. High or Medium priority) are not known. During the review of its application an applicant should monitor updates to NUREG-Q933 to identify any Note 4 issues that are subsequently prioritized and that meet the criteria for review specified above.

The criteria proposed by NEI in its October 21 letter did not include evaluation of the Note 2 issues for license renewal. For Note 2 issues, although the NRC may have developed a proposed resolution, action by the NRC to implement that resolution is not complete. Until final implementation requirements are promulgated, those Note 2 GSIs which contain aging issues should be evaluated for license renewal.

The staff's criteria implement the Commission's position, as discussed in the statements of consideration (SOC) for the amended license renewal rule (60 FR 22484), that a renewal applicant must evaluate aging issues associated with unresolved GSIs. However, the Commission also clearly stated that an aging management review involving an issue being addressed by the NRC as a GSI should not delay the issuance of a renewed license pending resolution of the issue. Four approaches were given in the SOC for addressing unresolved GSIs which the BTP also discusses. Application of the preceding criteria assures that all applicable unresolved GSIs are evaluated. The Low priority GSIs listed in NUREG-0933 need not be reviewed because, as stated in the introduction to NUREG-0933, these issues have been eliminated from further pursuit by the staff because the generic issues prioritization process has determined that there is little or no prospect of safety improvements. However, the NRC staff does periodically review existing Low priority GSIs to determine whether there is any new information that would necessitate reassessment of the original prioritization evaluations.

NEI has provided guidance for reviewing GSIs in Section 1.5 of its "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," NEI 95-10, Revision 0, which the staff proposed to endorse in its Draft Regulatory Guide, DG-1047, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses." The staff has developed the additional guidance contained in Enclosure 1, which updates and expands on the guidance found in NEI 95-10 and the BTP. The staff will incorporate the revised guidance into the next revision of the BTP. NEI is requested to consider incorporating the Enclosure 1 guidance into NEI 95-10.

To illustrate the application of the criteria contained in Enclosure 1, the staff reviewed NUREG-0933, updated through Supplement 21, December 1996, to identify GSIs involving potential aging effects that should be evaluated in a license renewal application. The results of the Staff's review are provided in Enclosure 2. An applicant is expected to apply the criteria contained in Enclosure 1 when preparing its application to review the version of NUREG-0933 in effect to identify GSIs needing evaluation.

NEI is requested to review the enclosures and inform the staff in writing of its intent regarding incorporating the Enclosure I guidance into NEI 95-10 and whether it agrees with the staff's application of the criteria to identify the GSIs listed in Enclosure 2.

If there are any questions. please contact Stephen Hoffman of my staff at 301-415-3245.

Sincerely,

Christopher I. Grimes, Project Director
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No 690

Enclosures: As stated

cc w/encl: See next page

B. Doroshuk, BGE
R. Gill, Duke
C. Pierce, SNC

Evaluation of USIs and GSIs for License Renewal
Additional Guidance for NEI 95-10. Section 1.5. Revision 0

INSERT LINE 9

Issues designated as unresolved safety issues (USIs) and generic safety issues (GSIs) are identified in the NRC program for the resolution of generic issues described in NUREG-0933, "A Prioritization of Generic Safety Issues." Appendix B to NUREG-0933 contains a listing of those issues that are applicable to operating and future plants. For license renewal, an applicant should evaluate all issues identified in NUREG-0933, Appendix B, with a Safety Priority/Status of USI, High, Medium, and Notes 1 and 2. The version of NUREG-0933 that is current on the date 6 months before the date of the application should be used for preparing the renewal application.

During the review of its application, an applicant should continue to review updates to NUREG-0933, Appendix B, to identify new issues added to NUREG-0933 that meet the preceding criteria and issues originally classified as Note 4, "Issue to be Prioritized in the Future," that are subsequently prioritized and that meet the preceding criteria for evaluation. An applicant should supplement its application to evaluate any new applicable issues identified up to 6 months before the anticipated date for issuing a renewed license.

Those issues that involve aging effects or time-limited aging evaluations for systems, structures, or components within the scope of license renewal should be specifically evaluated in the license renewal application. If, during the preparation or review of the application, the applicant or the NRC staff identifies an issue with a "Low" or "Drop" priority that involves potential aging effects and has specific applicability to the applicant's plant, the applicant should evaluate the issue in its application.

Insert Line 31

Normally, if resolution of a USI or GSI has been achieved before issuance of the renewed license, the first approach listed above, to incorporate implementation of that resolution into the application, should be followed.

Enclosure 2

Safety Issues Requiring Review
for License Renewal

ISSUE	SAFETY PRIORITY/ STATUS	TITLE
GSI 23	High	Reactor Coolant Pump Seal Failures
GSI 78	Medium	Monitoring of Fatigue Transient Limits for [Reactor Coolant System]
GSI 166	Note 1	Adequacy of Fatigue Life of Metal Components
GSI 168	Note 1	Environmental Qualification of Electrical Equipment
GSI 173.A	Note 2	Spent Fuel Storage Pool: Operating Facilities

REFERENCE 12

LICENSE RENEWAL ISSUE NO. 98-0082, SCOPING GUIDANCE, Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated August 5, 1999

August 5, 1999

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0082, SCOPING GUIDANCE

Dear Mr. Walters:

Enclosed is the staff's evaluation and proposed resolution for the subject issue. This item was originally limited to scoping concerns relating to "cascading," but has since been expanded to include guidance for the overall implementation of the requirements under 10 CFR 54.4. Changes to the draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plant" (SRP-LR) will be initiated to reflect this position. If there are any industry comments on the evaluations or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. We also expect that NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," be revised to reflect any necessary guidance to implement the attached staff position. If you have any questions regarding this matter, please contact Robert Prato at 301-415-1147.

Sincerely,

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure: As stated
cc w/enclosure: See next page

LICENSE RENEWAL ISSUE NO. 98-0082
SCOPING GUIDANCE

BACKGROUND

On the basis of the experience gained with the first two license renewal applications and comments provided on the standard review plan, the staff has identified the need for additional guidance for determining the systems, structures, and components (SSCs) within the scope of the rule. Specifically, the staff has identified concerns with the following scoping activities:

4. Determining the events that need to be considered for identifying the safety-related systems, structures, and components which are relied upon to remain functional during design-basis events. After the events have been identified, determining the SSCs relied upon to remain functional during and following these events to meet the criteria under 10 CFR 54.4(a)(1).

5. Determining the extent to which an applicant needs to apply "hypothetical failures" in identifying the SSCs under 10 CFR 54.4(a)(2) whose failure could prevent satisfactory accomplishment of any function required pursuant to 10 CFR 54.4(a)(1).

6. Determining the extent to which an applicant needs to "cascade" to the second-, third-, and fourth-level support systems in identifying the SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 54.4(a)(3).

The following specific requirements from the *Code of Federal Regulations,* and the Commission guidance from the Statements of Consideration published on May 8, 1995, in the *Federal Register* (60 FR 22461) apply to the scoping activities in question:

10 CFR 54.3 Definitions.

> *Current licensing basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR Parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant-specific design-basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.*

10 CFR 54.4 Scope.

(a) Plant systems, structures, and components within the scope of this part are --

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions --

i. The integrity of the reactor coolant pressure boundary;

ii The capability to shut down the reactor and maintain it in a safe shutdown condition; or

iii The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

The intended functions that these systems, structures, and components must be shown to fulfill in §54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1)-(3) of this section.

10 CFR 50.49(b)(ii)

Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events and natural phenomena for which the plant must be designed to ensure functions (b)(1)(i) (A) through (C).

SOC - Subsection III.e(i) Current Licensing Basis (60FR22465)

As defined in § 54.3 of the rule, the CLB is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and are in effect. A detailed explanation of the CLB, the regulatory processes underlying the CLB, compliance with the CLB, and consideration of the CLB is contained in the SOC for the previous license renewal rule (56 FR 64949:

December 13, 1991). In summary, the conclusions made in the SOC for the previous rule remain valid. The CLB represents the evolving set of requirements and commitments for a specific plant that are modified as necessary over the life of a plant to ensure continuation of an adequate level of safety. The regulatory process is the means by which the Commission continually assesses the adequacy of and compliance with the CLB. Compilation of the CLB is unnecessary to perform a license renewal review.

SOC - Subsection III.c(iii) Bounding the Scope of Review (60FR22467)

Pre-application rule implementation has indicated that the description of systems, structures, and components subject to review for license renewal could be broadly interpreted and result in an unnecessary expansion of the review. To limit this possibility for the scoping category relating to nonsafety-related systems, structures, and components, the Commission intends this nonsafety-related category (§54.4(a)(2)) to apply to systems, structures, and components whose failure would prevent the accomplishment of an intended function of a safety-related system, structure, and component. An applicant for license renewal should rely on the plant's CLB, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those nonsafety-related systems, structures, and components that are the initial focus of the license renewal review. Consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required.

Likewise, to limit the potential for unnecessary expansion of the review for the scoping category concerning those systems, structures, and components whose function is relied upon in certain plant safety analyses to demonstrate compliance with the Commission regulations (i.e., environmental qualification, station blackout, anticipated transient without scram, pressurized thermal shock, and fire protection), the Commission intends that this scoping category include all systems, structures, and components whose function is relied upon to demonstrate compliance with these Commission's regulations. An applicant for license renewal should rely on the plant's current licensing bases, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those systems, structures, and components that are the initial focus of the license renewal review. Consideration of hypothetical failures that could result from system interdependencies, that are not part of the current licensing bases and that have not been previously experienced is not required.

Several commenters noted that the word "directly" did not precede the phrase "prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section" in §54.4(a)(2) and concluded that, in the absence of the word "directly," the license renewal review could cascade into a review of second-, third-, or fourth-level support systems. The Commission reaffirms its position that consideration of hypothetical failures that could result from system interdependencies that <u>are not part of the CLB</u> and that have not been previously experienced is not required. However, for some license renewal applicants, the Commission cannot exclude the possibility that hypothetical failures that <u>are part of the CLB</u> may require consideration of second-, third-, or fourth-level support systems. In these cases the word "directly" may cause additional confusion, not clarity, regarding the systems, structures, and components required to be within the scope of license renewal. In removing the word "directly" from this scoping criterion, the Commission believes it has (1) achieved greater consistency *between the scope of the license renewal rule and the scope of the maintenance rule (§50.65) regarding nonsafety-related systems whose failure could prevent satisfactory accomplishment of safety-related functions and thus (2) promoted greater efficiency and predictability in the license renewal scoping process.*

The inclusion of nonsafety-related systems, structures, and components whose failure could prevent other systems, structures, and components from accomplishing a safety function is intended to provide protection against safety function failure in cases where the safety-related structure or component is not itself impaired by age-related degradation but is vulnerable to failure from the failure of another structure or component that may be so impaired. Although it may be considered outside the scope of the maintenance rule, the Commission intends to include equipment that is not seismically qualified located near seismically qualified equipment (i.e., Seismic II/I equipment already identified in a plant CLB) in this set of nonsafety-related systems, structures and components.

EVALUATION

1 Safety Related Systems Structures and Components - 10 CFR 54.4(a)(1)

In general, the scoping criteria under 10 CFR 54.4(a)(1) for safety-related SSCs are consistent with the criteria used by most licensees in defining safety-related SSCs under their current licensing bases. However, a number of the earlier licensed plants were not licensed to these criteria. These older plants were built and licensed to safety-related criteria that were based on identifying SSCs that served as barriers to the release of fission products. These licensees have been working to convert to the more current safety-related criteria for consistency across the industry, however, some differences may still exist.

Regardless of the criteria used by a licensee under 10 CFR Part 50, an applicant for license renewal needs to use the scoping criteria under 10 CFR 54.4(a)(1) for determining the safety-related SSCs within the scope of license renewal. However, these criteria need to be applied consistent with the plant's CLB. When an applicant's definition of safety-related SSCs within its CLB is not wholely consistent with the scoping criteria under 10 CFR 54.4(a)(1) for safety-related SSCs, an applicant needs to ensure that its scoping methodology clearly uses the criteria under 10 CFR 54.4(a)(1), and not its definition for safety-related SSCs. The following guidance can be used to determine the safety-related SSCs within the scope of license renewal:

- As stated in the scoping criteria under 10 CFR 54.4(a)(1), and referenced under 10 CFR 50.49(b)(1), to determine the scope of safety-related SSCs within the scope of license renewal, an applicant needs to consider those SSCs that are relied upon to remain functional during and following design-basis events, which are conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure the following functions:

  (i) The integrity of the reactor coolant pressure boundary;

  (ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

- To provide consistency with its current licensing basis (as defined 10 CFR 54.3) an applicant needs to consider *"the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect."* The NRC requirements referred to in this definition are the *"requirements contained in 10 CFR Parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100, and appendices thereto; orders, license conditions, exemptions, and technical specifications."*

It is expected that an applicant can limit its scoping review for safety-related SSCs to the design as documented in its plant-specific design basis information documents (e.g., the UFSAR) and written commitments that are docketed and in effect. In addition, an applicant may have to consider engineering evaluations, codes and standards, referenced topical reports, NUREGs and other documentation that serve an integral part of the applicant's compliance with and operation within the requirements and the commitments that are docketed and in effect. For example, if an applicant had a license condition to evaluate pipe breaks as part of its licensing basis, the applicant needs to consider its documented resolution if it is still in effect. In addition, the applicant may need to consider other documentation such as detailed evaluations that are the basis for the documented commitments that are docketed and in effect. Applicants need not consider site-specific evaluations, codes and standards, topical reports, or any other information that is not documented in its licensing documents or written commitments that are docketed and in effect that do not contribute to the compliance with and operation within NRC requirements.

2 Nonsafety-related Systems Structures and Components - 10 CFR 54.4(a)(2)

The scoping criterion under 10 CFR 54.4(a)(2), in general, is intended to identify those nonsafety-related SSCs that support safety related functions. More specifically, this scoping criterion requires an applicant to identify all nonsafety-related SSCs whose failure could prevent satisfactory accomplishments of the applicable functions of the SSCs identified under 10 CFR 54.4(a)(1). The SOC (60FR22467), Section III.c (iii) contains a clarification of the Commission's intent for this requirement in the following statement:

The inclusion of nonsafety-related systems, structures, and components whose failure could prevent other systems, structures, and components from accomplishing a safety function is intended to provide protection against safety function failure in cases where the safety-related structure or component is not itself impaired by age-related *degradation but is vulnerable to failure from the failure of another structure or component that may be so impaired.*

In addition, the SOC, Section III.c (iii), provides the following guidance to assist an applicant in determining the extent to which failures need to be consider when applying this scoping criterion:

Consideration of hypothetical failures that could result from system interdependencies, that are not part of the current licensing bases and that have not been previously experienced is not required. . . . However, for some license renewal applicants, the Commission cannot exclude the possibility that hypothetical failures that <u>are part of the CLB</u> may require consideration of second-, third-, or fourth-level support systems.

Therefore, to satisfy the scoping criterion under 10 CFR 54.4(a)(2), an applicant needs to identify those nonsafety-related SSCs (including certain second-, third-, or fourth-level support systems) whose failure can prevent the satisfactory accomplishment of the safety-related function identified under 10 CFR 54.4(a)(1). In order to identify such systems, an applicant would consider those failures identified in 1) the documentation that makes up its CLB, 2) plant-specific operating experience, and 3) industry-wide operating experience that is specifically applicable to its facility. The applicant need not consider hypothetical failures that are not part of the CLB, and that have not been previously experienced.

In determining the nonsafety-related SSCs that are within the scope of the rule, an applicant, for example, needs to consider including such SSCs as the following: 1) the portion of a fire-protection system that supplies water to the refueling floor (even if not required by the FP Plan) that is relied upon in a design basis accident analysis as an alternate source of cooling water that can be used to mitigate the consequences from the loss of spent fuel pool cooling; 2) a nonsafety-related, non-seismically qualified building whose failure could result in the failure of a tank that is relied upon as an alternate source of cooling water needed to mitigate the consequences of a design basis event; and 3) a segment of nonsafety-related piping identified as a Seismic II/I component in the applicant's CLB.

On the basis of the staff's experience to date, it is important to clarify that the scoping criterion under 10 CFR 54.4(a)(2) specifically applies to those functions "identified in paragraphs (a)(1)(i), (ii), and (iii)" of 10 CFR 54.4. An applicant need not extend this requirement to the scoping criteria under 10 CFR 54.4(a)(3), as is discussed below.

3 Commission Regulation Systems Structures and Components - 10 CFR 54.4(a)(3)

The scoping criteria under 10 CFR 54.4(a)(3) states that an applicant must consider *"[a]ll systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the [specified] Commission regulations[.]"* In addition, the SOC, Section III.c(iii), states that the Commission intended to limit the potential for unnecessary expansion of the review for SSCs that meet the scoping criteria under 10 CFR 54.4(a)(3), and provides additional guidance that qualifies what is meant by *"those SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission regulations. . ."* in the following statement:

> *[T]he Commission intends that this [referring to 10 CFR 54.4(a)(3)] scoping category include all systems, structures, and components whose function is relied upon to demonstrate compliance with these Commission's regulations. An applicant for license renewal should rely on the plant's current licensing bases, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those systems, structures, and components that are the initial focus of license renewal.*

Therefore, all SSCs that are relied upon in the plant's CLB (as defined in 10 CFR 54.3), plant-specific experience, industry-wide experience (as appropriate) and existing engineering analysis to perform a function that demonstrates compliance with and operation within the Commission regulations identified under 10 CFR 54.4(a)(3) are required to be included within the scope of the rule. For example, if a nonsafety-related diesel generator is required for safe shutdown under the fire protection plan, the diesel generator and all SSCs specifically required for that diesel to comply with and operate within the Commission's regulations based on the applicant's design specifications for that diesel shall be included within the scope of license renewal under 10 CFR 54.4 (a)(3). This may include, but should not be limited to the cooling water system or systems required for operability, the diesel support pedestal, and any applicable power supply cable specifically required for safe shutdown in the event of a fire.

In addition, the last sentence of the second paragraph in the SOC, Section III.c (iii), provides the following guidance for limiting the application of the scoping criteria under 10 CFR 54.4(a)(3) as it applies to the use of hypothetical failures:

> *Consideration of hypothetical failures that could result from system interdependencies, that are not part of the current licensing bases and that have not been previously experienced is not required.*

The SOC does not provide any additional guidance relating to the use of hypothetical failures or the need to consider second-, third-, or fourth-level support systems for scoping under 10 CFR 54.4(a)(3). Therefore, in the absence of this guidance, an applicant need not consider hypothetical failures or second-, third-, or fourth-level support systems in determining the SSCs within the scope of the rule required by the applicable Commission regulations. For example, if a nonsafety-related diesel generator is only relied upon to remain functional to demonstrate compliance with the Commission regulations, an applicant may not need to consider the following SSCs: 1) an alternate / backup cooling water system, 2) the diesel generator non-seismically qualified building walls, or 3) an overhead segment of non-seismically qualified piping (in a Seismic II/I configuration). This guidance is not intended to exclude any support system (identified by an applicant's CLB, actual plant-specific experience, industry-wide experience, as applicable, or existing engineering evaluations) that is specifically required for compliance with or operation within applicable Commission regulation. For example, if a nonsafety-related diesel generator (required to demonstrate compliance with an applicable Commission regulation) specifically requires a second cooling system to cool the diesel generator Jacket Water Cooling System for the diesel to be operable then both cooling systems must be included with the scope of the rule under 10 CFR 54.4(a)(3).

RESOLUTION

To identify the SSCs within the scope of license renewal consistent with the scoping criteria under 10 CFR 54.4 (a), an applicant needs to consider the following:

1 The safety-related systems, structures, and components which are relied upon to remain functional during and following design basis events (which are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events and natural phenomena for which the plant is designed) to ensure the functions under 10 CFR 54.4(a)(1)(i), (ii), and (iii). The events to be considered need to be determined by NRC requirements and licensee written commitments that are docketed and in effect.

2 The nonsafety-related SSCs (including certain second-, third-, or fourth-level support systems) whose failure can prevent the satisfactory accomplishment of the safety-related function identified under 10 CFR 54.4(a)(1). In order to identify such SSCs, an applicant needs to consider those failures identified in the CLB and, to the extent that it is applicable and appropriate, any plant-specific or industry-wide operating experience that is specifically applicable to the facility.

3 The SSCs that are relied upon in the plant's CLB to demonstrate compliance with the Commission regulations identified under 10 CFR 54.4(a)(3). In doing so, an applicant needs to consider those SSCs required to comply with and operate within the Commission regulations based on the applicant's CLB. In determining the SSCs within the scope of the rule under 10 CFR 54.4(a)(3), an applicant also needs to consider, to the extent that it is applicable and appropriate, any plant-specific or industry-wide operating experience that is specifically applicable to the facility.

REFERENCE 13

LICENSE RENEWAL ISSUE NO. 98-0030 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL COMPONENTS, Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated May 19, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 19, 2000

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, NW., Suite 400
Washington, DC 20006-3708

SUBJECT: LICENSE RENEWAL ISSUE NO. 98-0030, "THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL COMPONENTS"

Dear Mr. Walters:

Enclosed is the NRC staff's evaluation and proposed resolution for the subject issue. The staff plans to incorporate the recommended resolution as part of the next revision to the draft Standard Review Plan for License Renewal. Accordingly, if there are any industry comments on the evaluation basis or the proposed resolution, we request that you document those comments within 30 days following your receipt of this letter, to ensure a timely resolution of this issue. We also would be willing to meet with industry representatives to discuss any comments you may have. If you have any questions regarding this matter, please contact Sam Lee at (301) 415-3109.

Sincerely,

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 690

Enclosure: As stated

cc w/encl: See next page

STAFF EVALUATION OF LICENSE RENEWAL ISSUE NO. 98-0030

"THERMAL AGING EMBRITTLEMENT OF

CAST AUSTENITIC STAINLESS STEEL COMPONENTS"

1.0 BACKGROUND

Some of the primary pressure boundary and reactor vessel internal (RVI) components in U.S. light-water reactors are constructed from a cast austenitic stainless steel (CASS) material per American Society of Mechanical Engineers (ASME) Section III Specification SA-351. Examples of structures constructed from this type of material include pump casings, valve bodies, primary system piping, and RVI components of various configurations. NRC-sponsored research at Argonne National Laboratory (ANL) has shown that aging of CASS at reactor operating temperatures of 280-350CC (536-662°F) can lead to changes in the mechanical properties of these materials, depending on the characteristics of the material and the environment to which the component is exposed. The effects of thermal aging on materials include increases in the tensile strength, hardness, and Charpy impact energy transition temperature, as well as decreases in ductility, fracture toughness, and impact strength (Refs. 1-6).

CASS components have a duplex microstructure consisting of austenite and ferrite phases. The ferrite phase improves the tensile strength, castability, weldability, and stress-corrosion cracking resistance of the material. Exposing these steels to elevated temperatures promotes the formation of additional phases within the ferrite, causing the increased tensile strength, decreased ductility, and reduced fracture toughness associated with thermal aging. This thermal embrittlement mechanism can be severe enough to make the material 'susceptible to low energy fracture if the ferrite forms a continuous phase surrounding the grain boundaries in the microstructure. This low energy fracture is characterized by cleavage of the ferrite and low energy grain boundary separation of the austenite. The degree of embrittlement strongly depends upon the amount and distribution of the ferrite phase within the microstructure.

Research at ANL has shown that the most important factors in determining the extent of thermal aging in CASS are the chemical composition of the steel, the casting method used to construct the component, the amount of ferrite in the microstructure, and the service history (time and temperature) of the component. The chemical element most influential to the thermal aging process in U.S. steels is molybdenum (Mo), which is added to the steel to promote the formation of ferrite in the microstructure. CASS with high levels of Mo shows a higher susceptibility to thermal aging than steels with low Mo levels. The casting process greatly influences the cast microstructure and is also an important factor in determining the extent of thermal embrittlement. CASS components in the nuclear industry are typically manufactured by centrifugal or static casting. Static castings tend to show more susceptibility

Enclosure

to thermal aging than centrifugal castings. Since it is the ferrite phase that undergoes the microstructural changes leading to thermal embrittlement, elevated ferrite content in the steel results in greater susceptibility to thermal aging. ANL studies have shown that room temperature impact energy decreases with aging time and eventually reaches the lowest level attainable for a given composition, or the saturation level for that composition. The service temperature of a component will affect the rate at which the material reaches this saturation limit. However, prior to saturation, increased service temperatures will increase the level of embrittlement in a material for a given exposure time.

2.0 EVALUATION

The staff has reviewed several industry submittals addressing thermal aging of CASS materials, including Electric Power Research Institute (EPRI) Technical Report 106092 (Ref. 7), the license renewal application from Baltimore Gas and Electric for the Calvert Cliffs plants (Ref. 8), and several topical reports from reactor owners groups (Refs. 9 to 11). Each of these submittals addresses thermal aging embrittlement of CASS in a different manner; the submittal by EPRI will be used as the benchmark for evaluation of the "industry position~.

This evaluation addresses the industry position outlined in EPRI Technical Report 106092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems." The stated objectives of that report are to: (1) propose screening criteria to determine if a specific component should be inspected due to its potential susceptibility to thermal aging, (2) provide data supporting the proposed screening criteria, and (3) propose an aging management program for those components potentially affected by -thermal aging. The report references data produced from the research performed at ANL (Ref. 1).

Screening Criteria

The screening criteria proposed in EPRI TR-106092 are applicable to all Class I reactor coolant system and primary pressure boundary components constructed from SA-351 Grade CF3, CF3A, CF8, CF8A, CF3M, CF3MA, or CF8M. The factors used in the screening criteria are the same as those described in the background section: Mo content, casting procedure, and ferrite content. In the review of this proposed screening criteria, the staff considered saturated lower bound J integral vs. crack depth (J-R) curves. J-R curves measure a material's resistance to stable ductile crack growth.

EPRI's proposed screening criteria essentially divide all CASS components into the six categories shown in Table 1. The high Mo steels are those that meet CF3M, CF3MA, or CF8M grade specifications while the low Mo steels are those that meet CF3, CF3A, CF8, or CF8A grade specifications. The ferrite levels may be either calculated or measured values.

The industry proposes that all components deemed as having a potentially significant reduction in fracture toughness due to thermal aging must be placed in an aging management program, as described later.

Table I: Proposed Thermal Aging Screening Criteria in EPRI TR 106092

Mo Content (Wt. %)	Casting Method	Ferrite Content	Significant of Thermal Aging
High (2.0 – 3.0)	Static	All	Potentially Significant
	Centrifugal	> 20%	Potentially Significant
		≤ 20%	Non-significant
Low (0.50 max)	Static	> 20%	Potentially Significant
		≤ 20%	Non-significant
	Centrifugal	All	Non-significant

Supporting Data

ANL has developed procedures for conservatively predicting the J-R curve behavior of aged CASS based on material chemistry information and/or service history (Refs. 2 and 6). These correlations were developed from 80 different compositions of cast stainless steel which were aged up to 58,000 hours at 350°C (662°F). As part of this research program several heats of SA 351 material were aged and tested in order to compare measured saturated J-R curves with the ANL predicted values. These heats included both commercial and laboratory heats as well as static and centrifugal castings. In addition to these heats tested by ANL, the ANL analysis included fracture toughness data from other sources (Westinghouse, EDF, Framatome, and EPRI). In all cases the ANL predicted J-R curves were accurate or conservative corn pared to the measured values.

These measured and predicted J-R curves are used in the EPRI report to justify the proposed screening criteria described above. A deformation J value of 255 kJ/m^2 (1450 in-lb/in^2) at a crack depth of 2.5 mm (0.1 in) was used to differentiate between a non-significant and a potentially significant reduction in fracture toughness for fully aged materials. Flaw tolerance evaluations described in Appendices A and B of EPRI TR-106092 show that a material toughness of 255 kJ/m^2 (1450 in-lb/in^2) adequately protects against a loss of structural integrity in cast austenitic stainless steel components. The staff finds that Appendices A and B of EPRI TR-106092 provide an acceptable justification that 255 kJ/m^2 is an acceptable screening value to use in differentiating between non-significant and a potentially significant reduction in fracture toughness of aged CASS components.

ANL *also developed saturated lower-bound J-R curves for use when the corn position of the steel* is unknown (Refs. 2 and 6). In these situations given the steel grade, the casting procedure, and a measured ferrite level, a saturated lower bound J-R curve can be evaluated. The staff compared the J(2.5) values taken from these saturated lower bound J-R curves as well as J(2.5) values from the actual fracture toughness tests conducted on the various heats of material with the screening value of 255 kJ/m^2 for the various categories defined by the screening criteria as follows:

(1) High Mo - Static Castings

For high Mo static castings with ferrite levels >15 percent, the saturated lower bound J(2.5) is 221 kJ/m^2. At ferrite levels between 10 and 15 percent, the saturated lower bound J(2.5) becomes 257 kJ/m^2, and at ferrite <10 percent, J(2.5) increases to 322 k.J/m^2. Heat L examined by ANL was a CF8M static casting with 19 percent ferrite. The actual fracture toughness data for this heat showed a J(2.5) value of approximately 250 kJ/m^2. CF8M static castings are more susceptible to thermal aging so that even at low ferrite levels the saturated fracture toughness is relatively low.

Based upon the cited J(2.5) levels for ferrite levels between 10 and 15 percent, the staff finds that high Mo static cast components with ferrite levels below 14 percent are not susceptible to thermal aging. The proposed screening criteria in the EPRI report finding all high Mo static cast components potentially susceptible to thermal aging, regardless of ferrite content, is conservative.

(2) High Mo - Centrifugal Castings

In high Mo centrifugal castings, the saturated lower bound J(2.5) value is 259 kJ/m^2 for ferrite >15 percent and 298 kJ/m^2 for ferrite levels between 10 and 15 percent. Heat 205 is a CF8M centrifugal casting with 21 percent ferrite. The measured J(2.5) value of this particular heat is approximately 500 kJ/m^2 which is well above the lower bound and the screening value for J(2.5). Based on this data, the staff finds that only those high Mo centrifugal cast components with >20 percent ferrite show a significant reduction in fracture toughness. Therefore, the proposed screening criteria is acceptable.

(3) Low Mo - Static Castings

Low Mo static castings with ferrite levels >15 percent have a saturated lower bound J(2.5) of 342 kJ/m^2. Ferrite levels between 10 and 15 percent for this material have a saturated lower bound J(2.5) value of 377 kJ/m^2. Heat 69, a CF3 static casting containing 21 percent ferrite has a J(2.5) value of 516 kJ/m^2 which is also well above the screening value for J(2.5). Based on this data, the staff finds that the use of a 20-percent ferrite to differentiate non-significant and potentially significant reductions in fracture toughness is acceptable.

(4) Low Mo - Centrifugal Castings

In the case of low Mo centrifugal castings with ferrite levels >15 percent, saturated lower bound J(2.5) values are 450 kJ/m^2. Heat P1 is a CF8 centrifugal casting with 18 percent ferrite. This heat shows actual J(2.5) data to be approximately 700 kJ/m^2 which is well above the screening value for J(2.5). Even at these ferrite levels, low Mo centrifugal castings show adequate toughness. Therefore, the proposed screening criteria of non-significant for low Mo centrifugal cast components is acceptable.

Aging Management Program

The EPRI report proposes the following aging management program:

> All components deemed as having a potentially significant reduction in fracture toughness due to thermal aging should be inspected in accordance with the plants' inservice inspection program. Any of these detected flaws would then be evaluated according to ASME Section XI IWB-3640 "Evaluation Procedures and Acceptance Criteria for Austenitic Piping." If the sized flaws do not meet the IWB-3640 acceptance criteria, the component must then be repaired and/or replaced. If the component is deemed to have a non-significant reduction in fracture toughness or the sized flaws meet the IWB-3640 flaw acceptance criteria, the component can continue to operate within the current licensing basis.

Current inspection requirements in Table IWB-2500-1 of Section XI of the ASME Code for CASS components are the following:

- Piping (Category B-J): Volumetric and surface examination of pressure-retaining welds for NPS ~ 4 in.; surface examination of pressure-retaining welds for NPS<4 in.

- Valve Bodies (Categories B-M-1 and B-M-2): Visual VT-3 examination of internal surfaces and volumetric examination of pressure-retaining welds for NPS ~ 4 in.; surface examination of pressure-retaining welds for NPS <4 in.

- Pump Casings (Categories B-L-1 and B-L-2): Visual VT-3 of internal surfaces and volumetric of welds

- RV Internals (Category B-N-3): Visual VT-3 of surfaces

The inspection requirements for piping, valve bodies and pump casings are not a 100 percent inspection but rather an inspection of samples within each grouping.

The proposal in the EPRI report provides for inservice inspections in accordance with the plants' inservice inspection program. The staff does not think that this is adequate since components which may be susceptible to thermal aging, such as piping base metal and RV internals, are not currently covered to a sufficient degree by ASME Code requirements.

The NRC has previously approved ASME Section XI IWB-3640 for evaluating flaws in thermally aged cast stainless steel components for license renewal (Ref. 13). IWB-3640 procedures were developed from fracture toughness data of Types 316 and 304 welds. CF8M shows the greatest susceptibility to thermal aging of any of the other SA-351 grades considered in the screening criteria. A comparison of IWB-3640 weld data to the CF8M saturated lower bound curves shows that the toughness levels of these two materials are similar (Ref. 13). IWB-3641 (b) states that "[t]he evaluation procedures and acceptance criteria are applicable to.. .cast stainless steel (with ferrite level less than 20 percent)." However, the lower bound curve developed by ANL which was compared to the IWB-3640 submerged arc weld (SAW) data was for CF8M steels with 15-

25 percent ferrite. Based on the similarity of the fracture toughness data, the staff believes that IWB-3640 procedures would be applicable to thermally aged CASS with ferrite levels up to 25 percent.

The staff noted several limitations based on the ANL research regarding the development and use of their correlations:

- The ANL database used to develop these correlations had a maximum δ-ferrite content of 25 percent. Recent data (Ref. 1) has shown that applying these correlations to steels with ferrite levels in excess of 25 percent can result in a non-conservative overestimation of the actual fracture toughness of the material.
- Little data exists for centrifugal castings constructed from a high Mo grade of stainless steel.
- The ANL correlations were based on calculated ferrite levels using Hull's Equivalent Factors. Other procedures for calculating ferrite content may result in a non-conservative estimation of the fracture toughness of the steel.
- Niobium (Nb) increases CASS susceptibility to thermal aging. Since the ANL heats did not contain Nb, the correlations and screening criteria would not strictly apply to Nb-containing steels. This should not be an issue since the CASS components in U.S. light-water reactors do not contain Nb.

3.0 RESOLUTION

Based upon the review of the various industry submittals, the staff has developed the following position for management during the license renewal period of thermal aging in reactor components constructed of cast austenitic stainless steel (CASS).

Susceptibility Screening Method

Determination of the susceptibility of CASS components to thermal aging can use a screening method based upon the Mo content, casting method, and ferrite content. (Alternatively, components can be assumed as "potentially susceptible" without considering such screening.) The specific screening criteria acceptable to the staff are outlined in Table 2, and are applicable to all primary pressure boundary and reactor vessel internal (RVI) components constructed from SA-351 Grade CF3, CF3A, CF8, CF8A, CF3M, CF3MA, or CF8M, with service conditions above 250^{0}C (482^{0}F).

Table 2: CASS Thermal Aging Susceptibility Screening Criteria

Mo Content (Wt. %)	Casting Method	-Ferrite Level	Susceptibility Determination
High (2.0 – 3.0)	Static	≤ 14%	Not susceptible
		> 14%	Potentially susceptible
	Centrifugal	≤ 20%	Not susceptible
		>20%	Potentially susceptible
Low (0.50 max)	Static	≤ 20%	Not susceptible
		> 20%	Potentially susceptible
	Centrifugal	All	Not susceptible

Note that calculated &ferrite should use Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values).

The significance of finding a particular component not susceptible or potentially susceptible is described below for each component type. The examination requirements for each component type are provided in Table 3. In addition, acceptable flaw evaluation procedures are described.

Table 3: Examination Requirements for CASS Components

Component	Grouping	Not Susceptible	Potentially Susceptible
Piping (Base Metal)	NPS $\geq$ 4 in.	None	Inspection or evaluation
	NPS < 4 in.	None	Inspection or evaluation
Valve Bodies (Base Metal)	NPS $\geq$ 4 in.	ASME Section XI requirements	ASME Section XI requirements
	NPS < 4 in.	ASME Section XI requirements	ASME Section XI requirements
Pump Casings (Base Metal)	NPS $\geq$ 4 in.	ASME Section XI requirements	ASME Section XI requirements
	NPS < 4 in.	ASME Section XI requirements	ASME Section XI requirements
RV Internals (Base Metal)	Fluence $\geq 1 \times 10^{17}$	All: Supplemental examination or component-specific evaluation	
	Fluence $< 1 \times 10^{17}$	ASME Section XI requirements	Supplemental examination

Piping (Base Metal)

Since the base metal of piping does not receive periodic inspection in accordance with Section XI of the ASME Code, the susceptibility of piping constructed from CASS should be assessed for each heat of material. Alternatively, an assumption of "potentially susceptible" can be assumed for each heat or specific heats.

Should a particular heat be found "not susceptible," no additional inspections or evaluations are required to demonstrate that the material has adequate toughness.

Should a particular heat be found or assumed "potentially susceptible" and subject to plausible degradation (e.g., thermal fatigue), aging management can be accomplished through volumetric examination or plant/component-specific flaw tolerance evaluation. The volumetric examination should be performed on the base material of each heat, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress level, operating time and environmental considerations. Alternatively, a plant/component-specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate, toughness.

Valve Bodies and Pumo Casings

Valve bodies and pump casings are adequately covered by existing inspection requirements in
Section XI of the ASME Code, including the alternative requirements of ASME Code Case N-481 for pump casings. Screening for susceptibility to thermal aging is not required and the current ASME Code inspection requirements are sufficient.

Regarding valve bodies with NPS less than 4 in., this position is supported by a bounding fracture analysis finding that valves within this range do not require additional inspection or evaluation to demonstrate that the material has adequate toughness, even for severe thermal embrittlement conditions. (See attachment.)

Reactor Vessel Internals

For RVI components fabricated from CASS and hence subject to thermal embrittlement, concurrent exposure to high neutron fluence levels can result in a synergistic effect wherein the service-degraded fracture toughness is reduced from the levels predicted independently for either of the mechanisms. Therefore, components determined to be subject to thermal embrittlement require an additional consideration of the neutron fluence of the component to determine the full range of degradation mechanisms applicable for the component.

To account for this synergistic loss of fracture toughness, a program should be implemented consisting of either a supplemental examination of the affected components as part of the applicant's 10-year 151 program during the license renewal term, or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. The scope of the supplemental inspection should cover portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (e.g., Mo content, a-ferrite content, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress level, operating time and environmental conditions).

The component-specific evaluation looks first at the neutron fluence of the component. If the neutron fluence is greater than I x 1017 n/cm^2 (E> I MeV), a mechanical loading assessment would be conducted for the component. This assessment will determine the maximum tensile loading on the component during ASME Code Level A, B, C and 0 conditions. If the loading is compressive or low enough to preclude fracture of the component, then the component would not require supplemental inspection. Failure to meet this criterion would require continued use of the supplemental inspection program.

If the neutron fluence is less than I x 1017 n/cm^2 (E> I MeV), an assessment would be made to determine if the affected component(s) are bounded by the screening criteria in Table 2. In order to demonstrate that the screening criteria are applicable to RVI components, a flaw tolerance evaluation specific to the reactor vessel internals would be required, similar to that provided in Ref. 7. If the material is determined to be "potentially susceptible," then a supplemental examination would be required on the portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (e.g., Mo content, a-ferrite content, casting process, and operating temperature), and cracking susceptibility (applied stress level, operating time and environmental conditions). If the material is determined to be "non-

susceptible," no inspections or evaluations are required to demonstrate that the material has adequate toughness.

Supplemental Examination

The supplemental examination technique should be specified by the applicant in the license renewal application. Particular consideration must address the reliability of the supplemental examination technique in detecting the features of interest (such as crack appearance and size) in assuring the integrity of the component.

One example of a supplemental examination could be an enhancement of the visual VT-1 examination described in IWA-2210 of Section XI of the ASME Code. A description of such an enhanced VT-I examination could include the following characteristics: the ability to achieve a 1/2-mu (0.0005 in.) resolution, with the conditions (e.g., lighting and surface cleanliness) for the in-service examination bounded by those used to demonstrate the resolution of the inspection technique.

Volumetric Examination

Current volumetric examination methods are not adequate for reliable detection of cracks in CASS components. Should an acceptable method for volumetric examination of CASS components be developed, the performance of the equipment and techniques should be demonstrated through a program consistent with the ASME Code, Section XI, Appendix VIII.

Flaw Evaluation

Flaws detected in CASS components should be evaluated in accordance with the applicable procedures of IWB-3500 in Section XI of the ASME Code. If the a-ferrite content does not exceed 25 percent, then flaw evaluation would be in accordance with the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20 percent in IWB-3641 (b)(I). If the CASS material is "potentially susceptible" and the a-ferrite content exceeds 25 percent, then flaw evaluation would be on a case-by-case basis using fracture toughness data supplied by the licensee, such as that published by Jayet-Gendrot, et al (Ref. 14).

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 6, 1999

MEMORANDUM TO: William H. Bateman, Chief
Chemical and Materials Engineering Branch
Division of Engineering, NRR

FROM: Michael E. Mayfield, Chief
Materials Engineering Branch
Division of Engineering Technology, RES

SUBJECT: INTEGRITY OF <4-INCH NPS VALVE BODIES MADE FROM CAST STAINLESS STEEL

Based on recent discussions among the staff, an issue was identified relating to the potential for thermal aging to degrade the integrity of valve bodies made from cast duplex stainless steels, commonly referred to as simply Cast stainless steels or CASS. The concern was specific to those valve bodies with a high delta-ferrite content. The issue focused on 4-inch NPS and smaller valves because periodic in-service inspections would identify cracking in larger valve bodies before they could propagate to a critical size. However, in-service inspection for the valves less than 4-inch NPS does not require internal visual or volumetric inspection of the valve bodies. Rather, the ASME Code (Section XI, 1995 Edition, Table IWB2500-1) requires surface examination of essentially 100 percent of all welds for at least one valve within each group of valves that are of the same size, constructional design, and manufacturing method, and that perform similar functions in the system. Thus, the staff was considering the need for additional inspection or evaluation criteria for these small valves.

The Materials Engineering Branch staff undertook two activities to evaluate the need for additional guidance. First, we reviewed the Licensee Event Report database and the Nuclear Reliability Data System (NPRDS) database to identify instances of cracking of valve bodies. Secondly, we performed a conservative bounding integrity analysis to estimate the crack sizes that could be present in degraded valve bodies without challenging the integrity of the valve.

Based on the information discussed below, we found that (1) there have been no reported instances of valve body cracking in these smaller size valves made from cast stainless steel, and (2) aged CASS valve bodies, even with extremely low fracture toughness, can withstand very large through-wall cracks.

Regarding the review of the LER and NPRDS databases, Attachment I provides a summary of the event reports which identified valve body cracks. The search included the Sequence Coding and Search System (SCSS), the NPRDS, and foreign event files for thermal fatigue. The SCSS search covered the last 20 years, and the NPRDS search covered 1987 to 1996.

Attachment

Ten events were identified but none of them involved small diameter CASS valve bodies. Six of the events were for valves less than 4-inch NPS, but none of those events were for CASS materials. Two of the events were associated with CASS material but were for 8-inch and 24-inch valves. Thus, service experience does not suggest a significant degradation mechanism for CASS valve bodies. While service experience alone does not provide a basis to eliminate the staff's concern, it also does not suggest that these valves are particularly susceptible to service cracking, a necessary prerequisite to a loss of integrity of the component.

With regard to the bounding integrity analysis, Attachment 2 provides information concerning the details of the analysis. An elastic-plastic assessment was performed using the uR6~ Failure Assessment Diagram (FAD) methodology. This method has been shown to provide conservative assessment of the fracture integrity of operating structures. While the fracture mechanics formulation is specifically for cracks in a flat plate rather than a valve body, the overall bounding nature of the analysis is believed to offset this factor.

The key inputs to the analysis are the stress in the valve body, the yield and tensile strength of the material, Young's modulus, and the fracture toughness of the aged material. The stress values were obtained from earlier work performed by INEEL for another project addressing erosion-corrosion of valve bodies. In that work, a finite element analysis was performed for a 16-inch globe valve in the normally closed position. Full system pressure (225 psig for this valve) was applied to one side of the valve, in addition to seismic stresses and end-moments from the piping system analysis. In one computer run, the most severely eroded areas were modeled with a minimum wall thickness of 0.10-in, versus the 0.5 - 0.8-in, wall thickness actually observed in the valve. The peak stress found in the most severely eroded areas under these conditions varied between 22.9 ksi and 41.4 ksi. Yield stress at the applicable temperature is 34.4 ksi. It is important to note that even though the model simulated more severe erosion than was actually observed, these higher stresses only occurred in very small areas of the valve body. Displacements were sufficiently small so that the operation of the valve was judged not to be compromised. Stresses under normal operating pressures in areas that had not been eroded were significantly lower. For these reasons, we chose to use a .stress of 20 ksi in the current fracture analysis. While the INEEL stress analysis is not specific to small diameter valves, it is believed to represent a high-stress condition for valve bodies and was used as input to this bounding analysis.

The yield strength, tensile strength, and Young's modulus values were taken from the ASME Code SC II for 550 F operating temperature. These values are for unaged materials but were used in lieu of specific data on the aged values.

With regard to the fracture toughness value, the staff's concern was fOr situations where the CASS material had a high delta ferrite content, specifically greater than 25 percent. Our research program did not include materials with these very high values of delta ferrite so we did not have specific data from which we could provide a bounding estimate. Consequently, we contacted Dr. 0. Chopra of Argonne National Laboratory, who had performed our research in this area. Based on his knowledge of the literature, Dr. Chopra suggested a value as low as 69 ksi.in'A could be considered a worst-case fracture toughness for these materials.

With these conservative input assumptions, the FAD analysis shows that the small diameter CASS valve bodies could withstand a through-wall defect approximately 1.35 inches long. While the specific value would be specific to the application, the analysis demonstrates that the
CASS valve bodies are flaw-tolerant, even for severely aged materials.

Based on the fact that we did not identify any service failure history associated with these small diameter CASS valve bodies, and the fact that they can withstand very long through-wall cracks, even under high stresses, suggests that additional inspections during a license renewal period are not warranted. We therefore conclude that the present requirements for in-service inspection are adequate.

If you or your staff have questions concerning this analysis, please contact me at (301) 415-6690 or Mark Kirk at (301) 415-6015.

Attachments: As stated

ATTACHMENT I

SUMMARY OF VALVE BODY CRACKING EVENTS

The following event reports were identified with valve body cracks in a search of operational experience database files and several technical reports. The data search included the Sequence Coding and Search System (SCSS), the Nuclear Reliability Data System (NPRDS), and the foreign event files for thermal fatigue. The SCSS search covered the time period of the past 20 years, and the NPRDS search covered the period from 1987 to 1996. The technical reports included (1) INEL-95/0648, "An Evaluation of the Effects of Valve Body Erosion on MOV Operability," (2) NUREG/CR-4302, "Aging and Service Wear of Check Valves Used in ESF Systems of Nuclear Power Plant," (3) NUREG/CR-4747, "An Aging Failure Survey of LW Reactor Safety Systems and Components," and (4) NUREG/CR-6246, "Effects of Aging and Service Wear on Main steam Isolation Valves and Valve Operators."

I. Peach Bottom, LER 277196-004

A small leak in the HPCI cooling water line relief valve (lxl-1/2" Crosby, Model JMB-C-E). The failure mechanism was IGSCC. It was determined that the relief valve base material consisted of a nickel alloy which, due to a high carbon content (0.4%), is highly susceptible to IGSCC.

2. Indian Point 3. LER 286195-024

Both valves of SWN-43-5 and -43-1 of the essential service water containment isolation were found to be leaking through valve body. It was confirmed that the valve body had a small hole and UT had shown possible valve body wall thinning. The cause was under-deposit, oxygen concentration cell corrosion and/or microbiologically induced corrosion due to long-term stagnant service water. The valves were made of carbon steel and 2" size.

3. South Texas 1. LER 498188-22

Slight leakage occurred at a number of locations in the aluminum-bronze Essential Cooling Water (ECW) system. Further investigation revealed that some small bore (2 inch and smaller) fittings and valves in the ECW system have undergone extensive crevice corrosion, resulting in through wall seepage.

4. Salem 1. LER 272190-026

Through wall main steam (MS) leak at body of a check valve. This 1" Type 316 stainless steel valve (2MS57) was for the MS& turbine bypass AFW pump drain header. The failure was attributed to wall thinning due to erosion/corrosion.

5. Salem 2. LER 31 1188-22

Containment spray valves 21 & 22 revealed cracks in the valve castings. Visual examination revealed a 2.5" crack with a buildup of boric acid crystals. An analysis indicated the apparent cause was attributed to TGSCC. These valves were 8" stainless steel gate valve (SA-351, Grade CF8).

6. Duane Arnold I (event date: 0711411989)

The 'B' vent control valve (2" carbon steel) in the condensate demineralizer system was found leaking severely. The cause was flaw in casting of valve body and erosion.

7. Loviisa 2 (German) (event date: 1994)

A leakage was observed through the body of a control valve in a pressurizer auxiliary spray line. The valve body was forged titanium stabilized austenitic stainless steel and of 2" size. The cracking was considered to be caused by thermal stratification and mixing.

8. Haddam Neck. LER 213/96-019-01

A pinhole leak in the body of an 8" RHR isolation valve (RH-V-791A) to the "A" RHR heat exchanger. A small buildup of boric acid on the valve body was noted. The root cause was not determined. This was a stainless steel gate valve Model 2216-SP manufactured by Aloyco (Crane).

9. Palisades. LER 255194-006

An accumulation of boric acid on the valve body of 24" austenitic stainless steel (SA-351, Grade CF8M) check valve (CK-ES-3166) was confirmed to be caused by a through wall defect in the valve body. The valve is located between the containment sump and the suction piping for one train of the engineered safeguard system pumps. The cause was preferential corrosion at the grain boundary in a weld-repaired region of the valve casting.

10. Cooper. LER 298/93-014

A small through-wall leak was observed from a 18" valve in the SW line to the R.R. heat exchanger. The leak was determined to be caused by localized erosion. The valve is a 18" carbon steel Anchor Darling glove valve. Erosion of a large globe valve was the subject of NRC Information Notice 89-01.

Contact: Chuck Hsu (415-6356)

I

ATTACHMENT 2

Bounding Fracture Analysis of Inspection Requirements for Valve Bodies and Pump Casings having NPS , 4-in.

BACKGROUND

This attachment details the results of a bounding analysis on the fracture resistance of small diameter cast austenitic stainless steel (CASS) valve bodies, NPS '4-in, with high delta ferrite (>25 percent) after severe thermal embrittlement. This analysis was undertaken to help determine if licensees should be required to perform either (a) inspection, or (b) analysis to demonstrate the fracture integrity of these components during a license extension period.

METHODOLOGY

An elastic-plastic fracture assessment was performed according to the "R6" Failure Assessment Diagram methodology developed by the Central Electricity Generating Board in the United Kingdom [1,2]. Adherence to the protocols described in references [1,2] has repeatedly been demonstrated to prOvide conservative assessments of the fracture integrity of operating structures.

INPUTS

o Stress values were obtained from earlier work performed by INEEL for another project addressing erosion-corrosion of valve bodies [3]. In that work, a finite element analysis was performed for a 16-inch globe valve, in the normally closed position. Full system pressure (225 psig for this valve) was applied to one side of the valve, in addition to seismic stresses and end-moments from the piping system analysis. In one computer run, the most severely eroded areas were modeled with a minimum wall thickness of 0.10-in, versus the 0.5 - 0.8-in. wall thickness actually observed in the valve. The peak stress found in the most severely eroded areas under these conditions varied between 22.9 ksi and 41.4 ksi. Yield stress at the applicable temperature is 34.4 ksi. It is important to note that even though the model simulated more severe erosion than was actually observed, these higher stresses only occurred in very small areas of the valve body. Displacements were sufficiently small so that the operation of the valve was judged not to be compromised. Stresses under normal operating pressures in areas that had not been eroded were significantly lower. For these reasons, we chose to use a stress of 20 ksi in the current fracture analysis. While the INEEL stress analysis is not specific to small diameter valves, it is believed to represent a high-stress condition for valve bodies and was used as input to this bounding analysis.

- The following properties are taken from ASME Code SC II for use in this analysis.

  They are representative of CASS properties (SC 351-CF8) at 550°F without thermal embrittlement.
  - Yield strength 18 ksi
  - Ultimate strength: 67 ksi
  - Modulus: 25,550 ksi

- RES does not have specific fracture toughness test data for aged CASS materials with delta ferrite >25 percent. However, Dr. 0. Chopra *(Argonne National Laboratory)*, who performed the NRC's research on the fracture toughness of CASS materials, described work performed by EDF on both severely aged CASS (up to 100,000 hours), and trepan samples removed from operating components. From this work he suggested that the lowest observed J_{1-} value for CASS was on the order of 171 in-lbs/in^2 (30 *kJlm^2)*. This fracture toughness corresponds to a casting having a ferrite content of between 35% and 45%. This $J_{,0}$ was converted to an equivalent *K* value of 69 ksi*in$^{0.5}$ assuming plane strain conditions.

- A valve body thickness of %-in was assumed. However, because of the assumptions of the collapse solution (see below), valve body thickness does not enter the analysis.

FLAW MODEL / IDEALIZATION

As this was a bounding analysis, it was of interest to demonstrate that the valve body having the lowest anticipated toughness could sustain a through-wall crack in the presence of the highest anticipated stress without fracturing. The R6 methodology requires that both a stress intensity factor (K) solution and a collapse solution be available for the flaw in question. The *K* solution for a through-wall crack in an infinite body is as follows:

$$K= \sigma_{applied} \sqrt{\pi \alpha} \qquad (1)$$

where *a* is half of the through-wall crack length. For the collapse solution, it was assumed that the crack would not be large enough to significantly diminish the load-bearing cross section of the valve.

RESULTS

To perform an Option I R6 analysis, two quantities are computed: K, and L~. Kr is the ratio of the applied stress intensity factor (from eq. (1)) to the material fracture toughness (69 ksi*ino^5 in this case). L, is the ratio of the applied stress (20 ksi) to the yield stress (18 ksi). A point at location (Lr, K,) is then plotted on a general failure assessment diagram, as illustrated in Figure 1. On this diagram, points located between the axes and the failure assessment curve (a lower-bound curve appropriate to all metallic materials) are deemed to be "safe," while those outside of the failure assessment curve are "unsafe." The curve is thus a failure locus. In this analysis we increased the length of the crack (a in eq. (1)) until the assessment point lay on the curve. By this method, we determined that the CASS valve could sustain a 1.35-in long through wall crack before failure occurred.

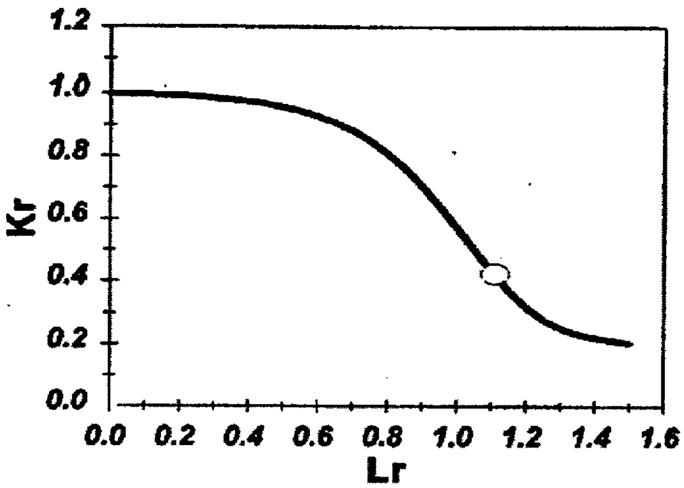

Figure 1: Failure Assessment Diagram.

CONCLUSIONS

Even after severe thermal embrittlement, a CASS valve loaded to the maximum anticipated stress can sustain a through wall crack well in excess of its wall thickness without fracturing. The worst case conditions assumed here suggest that requirements for licensees to either (a) inspect, or (b) provide analysis to demonstrate the fracture integrity of these components would represent an unnecessary duplication of effort.

REFERENCES

[1] Milne, I., et al., "Assessment of the Integrity of Structures Containing Defects," CEGB Report R/H/R6 (Revision 3), 1986.

[2] Mime, in., et al., "Assessment of the Integrity of Structures Containing Defects," *mt. J. Pres. Ves. & Piping,* 32 (1988), 3-104.

[3] Hunt, T. H. and Nitzel, M. E., "An Evaluation of the Effects of Valve Body Erosion on Motor-Operated Valve Operability," INEL-95/0648, December 1995.

REFERENCE 14

"ST. LUCIE, UNITS 1 AND 2, EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 54, SECTION 54.21(b) REGARDING SCHEDULE FOR SUBMITTING AMENDMENTS TO THE LICENSE RENEWAL APPLICATION (TAC NOS. MB3406 AND MB3412)", Letter to J. A. Stall, Florida Power and Light Company, from Noel F. Dudley, NRC, dated November 19, 2002

November 19, 2002

Mr. J. A. Stall
Senior Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE, UNITS 1 AND 2, EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 54, SECTION 54.21(b) REGARDING SCHEDULE FOR SUBMITTING AMENDMENTS TO THE LICENSE RENEWAL APPLICATION (TAC NOS. MB3406 AND MB3412)

Dear Mr. Stall:

The Commission has approved the enclosed exemption from the specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, Section 54.21(b), for St. Lucie, Units 1 and 2.

A copy of the exemption and the supporting safety evaluation are enclosed. The exemption has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Noel F. Dudley, Senior Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-335 and 50-389

Enclosures: 1. Exemption
2. Safety Evaluation

cc w/encls: See next page

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE, UNITS 1 AND 2

DOCKET NOS. 50-335 AND 50-389

EXEMPTION

1.0 BACKGROUND

The Florida Power and Light Company, et al. (FPL, the applicant) is the holder of Facility Operating License Nos. DPR-67 and No. NPF-16, which authorize operation of St. Lucie, Units 1 and 2, respectively. The licenses provide, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of two pressurized water reactors located in St. Lucie County, Florida.

2.0 REQUEST/ACTION

Title 10 of the Code of Federal Regulations (10 CFR), Part 54 addresses the various requirements for renewal of operating licenses for nuclear power plants. Section 54.21(b) of 10 CFR specifies:

> Each year following submittal of the license renewal application and at least 3 months before scheduled completion of the NRC review, an amendment to the renewal application must be submitted that identifies any change to the CLB [current licensing basis] of the facility that materially affects the contents of the license renewal application, including the FSAR [final safety analysis report] supplement.

ENCLOSURE 1

In accordance with 10 CFR 54.15, which references 10 CFR 50.12, the NRC staff, upon its own initiative, developed an exemption to 10 CFR 54.21(b) for St. Lucie, Units 1 and 2. At the time that 10 CFR Part 54 was issued, the staff expected that its review of a license renewal application (LRA) could take three or more years. The NRC staff completed its reviews of recent LRAs in less than 20 months. The exemption would allow FPL to submit one LRA amendment during the staff's review of the application, instead of two amendments.

The NRC staff anticipates completing its review of the St. Lucie, Units 1 and 2, LRA and issuing a safety evaluation report (SER) by July 3, 2003. This exemption would permit FPL to forgo submitting an annual LRA amendment provided it submits a single LRA amendment for St. Lucie, Units 1 and 2, at least three months before this scheduled completion date.

3.0 DISCUSSION

Pursuant to 10 CFR 54.15, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 54, in accordance with the provisions of 10 CFR 50.12, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present.

The requirements for exemption are discussed below:

The Commission's basis for requiring applicants to submit amendments to LRAs is contained in Section 54.21(b) and is discussed in the 1991 Statements of Consideration for Part 54 (56 FR 64954). The Commission established the requirement to ensure that the effects of changes to the renewal applicant's CLB is evaluated during the review of its renewal application. The exemption is consistent with the Commission's intent for the NRC staff, during its review of the application, to evaluate changes to the CLB of the facility that materially affects the contents of the LRA, including the FSAR supplement.

The exemption seeks only schedular relief regarding the timing and number of amendment submittals, and not substantive relief from the requirements of Parts 50, 51, or 54. FPL must still submit an LRA amendment for St. Lucie, Units 1 and 2, as required by 10 CFR Part 54. Therefore, the NRC staff finds that granting this schedular exemption will not represent an undue risk to public health and safety and is consistent with the common defense and security.

3.1 Special Circumstances Supporting Issuance of the Exemption

An exemption will not be granted unless special circumstances are present as defined in 10 CFR 50.12(a)(2). Specifically, Section 50.12(a)(2)(ii) states that a special circumstance exists when "Application of the regulation in the particular circumstances. . . is not necessary to achieve the underlying purpose of the rule. . ." In initially promulgating Section 54.21(b) in 1991, the Commission stated that the purpose of submitting LRA amendments is "To ensure that the effect of changes to a license renewal applicant's existing licensing basis is evaluated during the review of a renewal application, renewal applicants will be required to update the renewal application (including the integrated plant assessment) annually;" (56 FR 64954). The Commission indicated that the changes to the CLB that could affect the results of the license renewal processes, such as, scoping, screening, and aging management reviews should be evaluated during the NRC review of the LRA. As set forth below, the applicant's submittal of a single LRA amendment would allow the NRC staff to review and document the licensing changes in its safety evaluation report (SER) for St. Lucie, Units 1 and 2. Accordingly, under the exemption, the NRC staff will have the opportunity to review the recent changes to the CLB that could affect the results of license renewal processes.

The applicant submitted its LRA for St. Lucie, Units 1 and 2, to the NRC on November 29, 2001. The NRC staff is scheduled to complete its review and the SER by July 3, 2003. In accordance with the requirements of 10 CFR 54.21(b), an applicant must submit a yearly LRA amendment by November 29, 2002, and a second amendment before

April 3, 2003, which is three months before the NRC staff is scheduled to complete its review and issue an SER. Consequently, the licensee is required to submit two amendments within four months.

The SER with open items, which is scheduled to be issued by February 7, 2003, will identify proposed licensee commitments that change the CLB and are acceptable to the NRC. The applicant will be able to include these changes in an amendment that is submitted after the SER with open items is issued. The NRC staff can then review these changes and revise the SER, accordingly. Hence, submittal of a single amendment after the SER with open items is issued would be beneficial to the NRC staff and the licensee.

Therefore, submittal of two LRA amendments to satisfy the intent of Section 54.21(b) and the application of the regulation, in this case, is not necessary to achieve the underlying purpose of the rule. The NRC staff finds that the exemption meets the requirement in Section 50.12(a)(2)(ii) that special circumstances exist to grant the exemption.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 54.15 and 10 CFR 50.12, the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The exemption allows the applicant to forgo submitting the annual LRA amendment provided it submits an LRA amendment at least three months before the scheduled completion of the NRC's review. Therefore, the Commission hereby grants FPL the proposed exemption from the requirements of 10 CFR 54.21(b) for St. Lucie, Units 1 and 2, based on the circumstances described herein.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (67 FR 69254).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 19th day of November, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
EXEMPTION FROM 10 CFR 54.21(b)
REGARDING SUBMITTAL OF AMENDMENTS TO THE
FLORIDA POWER & LIGHT COMPANY. ET AL.
ST. LUCIE. UNITS 1 AND 2
DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

Requirements for filing applications for renewed operating licenses are contained in the license renewal rule, Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54, Section 54.21(b), which states: "Each year following submittal of the license renewal application and at least 3 months before scheduled completion of the NRC review, an amendment to the renewal application must be submitted that identifies any change to the CLB [current licensing basis] of the facility that materially affects the contents of the license renewal application, including the FSAR [final safety analysis report] supplement."

The NRC staff, on its own initiative, proposed an exemption that would allow Florida Power and Light Company (FPL) to submit a single amendment at least three months before the NRC staff issues its safety evaluation report (SER) for the St. Lucie, Units 1 and 2, license renewal application (LRA). Such an exemption would allow FPL to identify recent CLB changes, which affect the results of license renewal processes, such as, scoping, screening, and aging management reviews, and submit the information in a single LRA amendment. The exemption provides efficiencies for both FPL and the NRC by reducing the number of amendments that are required to be submitted and reviewed.

2.0 EVALUATION

Section 54.15 of 10 CFR states that exemptions from the requirements of Part 54 may be granted by the Commission in accordance with Section 50.12. An exemption may be granted under Section 50.12 if the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. However, an exemption will not be granted unless special circumstances are present as defined in Section 50.12(a)(2).

The Commission's basis for establishing the requirement for submitting LRA amendments contained in Section 54.21(b) is discussed in the 1991 Statements of Consideration for Part 54 (56 FR 64954). The Commission established the requirement to ensure that the effect of changes to the renewal applicant's CLB is evaluated during the review of a renewal application.

FPL submitted its LRA for St. Lucie, Units 1 and 2, to the NRC on November 29, 2001. The NRC staff is scheduled to complete its review and issue the associated SER by July 3, 2003. In accordance with the requirements of 10 CFR 54.21(b), the applicant must submit a yearly LRA amendment by November 29, 2002, and a second amendment before April 3, 2003, which is three months before the NRC staff is scheduled to complete its review and issue the associated SER. Consequently, the licensee is required to submit two amendments within four months.

The Commission indicated in the 1991 Statements of Consideration for Part 54 (56 FR 64962), that the technical review would take approximately two years and that any necessary hearings

ENCLOSURE 2

could take an additional year or more. Hence, the review and approval of a LRA could take three years or more. In the context of a three-year review, the requirement to submit a yearly update allows the NRC staff sufficient time to review changes to the CLB. However, FPL has not yet made any change to the CLB that materially affects the contents of the LRA, including the FSAR supplement, that requires NRC staff evaluation. There are no hearings associated with the review of the St. Lucie, Units 1 and 2, LRA. FPL plans to make changes to the CLB after the NRC staff issues the SER with open items, which is scheduled to be completed by February 7, 2003.

Therefore, should the Commission determine to grant the exemption, the NRC staff would be able to evaluate the effects of changes to the CLB during its review of the LRA.

The exemption seeks only schedular relief regarding the number and dates of submittals, and not substantive relief from the requirements of Parts 50, 51, or 54. FPL must still submit an LRA amendment identifying any changes to the CLB of the facility that materially affects the content of the LRA, and the FSAR supplement. The NRC staff will verify that all applicable Commission regulations have been met before issuing the renewed licenses. Therefore, the NRC staff finds that granting this scheduler exemption will not represent an undue risk to public health and safety and granting the exemption is consistent with the common defense and security.

2.1 Special Circumstances Supporting Issuance of the Exemption

An exemption will not be granted unless special circumstances are present as defined in Section 50.12(a)(2). Specifically, Section 50.12(a)(2)(ii) states that a special circumstance exists when "Application of the regulation in the particular circumstances ... is not necessary to achieve the underlying purpose of the rule." In initially promulgating Section 54.21(b) in 1991, the Commission stated that the purpose of submitting LRA amendments is "To ensure that the effect of changes to a renewal applicant's CLB is evaluated during the review of a renewal application, renewal applicants will be required to update the renewal application (including the integrated plant assessment) annually." (56 FR 64954.)

At that time, the Commission indicated that the technical review would take approximately two years and any necessary hearings could take an additional year or more (56 FR 64962).

FPL submitted its LRA for St. Lucie, Units 1 and 2, to the NRC on November 29, 2001. The NRC staff is scheduled to complete its review and issue the associated SER by July 3, 2003. In accordance with the requirements of 10 CFR 54.21(b), the applicant must submit a yearly LRA amendment by November 29, 2002, and a second amendment before April 3, 2003, which is three months before the NRC staff is scheduled to complete its review and issue the SER. Consequently, the licensee is required to submit two amendments within four months.

The NRC staff is scheduled to issue an SER with open items for the St. Lucie, Units 1 and 2, LRA by February 7, 2003. In the SER with open items, the NRC staff will identify proposed changes to the CLB that are acceptable to the NRC staff. FPL plans to document and submit these proposed changes in an LRA amendment at least three months before the completion of the NRC staff review. The NRC staff can then review and confirm the adequacy of the information in the LRA amendment and include its review in the SER, which is scheduled to be completed by July 3, 2003.

Since it submitted the LRA, FPL has not made any change to the CLB that materially affects the contents of the SER, including the FSAR supplement. FPL plans to make changes to the CLB after the NRC staff issues the SER with open items. FPL plans to document these changes in an LRA amendment at least three months before the scheduled completion of the NRC staff review.

The requirement in 10 CFP 54.21(b) for submittal of yearly LRA amendments is based on an NRC staff review lasting over three years. The NRC staff is scheduled to complete its review of the St. Lucie, Units 1 and 2, LRA and issue the associated SER by July 3, 2003. The submittal of a single amendment three months prior to completing its review would allow the NRC staff to evaluate changes to the CLB and revise its SER. The exemption will reduce the burden on the applicant and will allow for a more efficient NRC staff review resulting from a single amendment being submitted after the SER with open items is issued.

Therefore, submittal of a single amendment would satisfy the intent of Section 54.21(b), and the application of the regulation in this case is not necessary to achieve the underlying purpose of the rule. The NRC staff finds that the exemption meets the requirement in Section 50.12(a)(2) that special circumstances exist to grant the exemption.

3.0 CONCLUSION

Based on the foregoing, the NRC staff finds that the exemption is acceptable in that it is authorized by law, will not present an undue risk to the public health and safety, is consistent with the common defense and security, and special circumstances are present under 10 CFR 50.12(a)(2)(ii). The exemption allows the applicant to forgo submitting the annual LRA amendment provided it submits an LRA amendment at least three months before the scheduled completion of the NRC's review. The application must identify any changes to the CLB of the facility that materially affects the contents of the LRA, including the FSAR supplement. In the course of its review of the LRA amendment for St. Lucie, Units 1 and 2, the NRC staff will evaluate the effects of changes to the renewal applicant's CLB.

Principal Contributor: Noel F. Dudley, NRR

Date: November 19, 2002

REFERENCE 15

"INDUSTRY RESPONSE – CONSOLIDATED LIST OF COMMITMENTS FOR LICENSE RENEWAL, DECEMBER 16, 2002," Letter to P.T. Kuo, NRC, from Alan Nelson, NEI, dated February 26, 2003

NUCLEAR ENERGY INSTITUTE

Alan P Nelson
SENIOR PROJECT MANAGER,
LICENSING
NUCLEAR GENERATION

February 26, 2003

Dr. P.T. Kuo
Program Director
License Renewal and Environmental Impacts
Division of Regulatory Improvement Programs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Industry Response — Consolidated List of Commitments for License Renewal, December 16, 2002

Dear Dr. Kuo:

On December 16, 2002, NEI received your correspondence requesting that each license renewal applicant provide the staff with consolidated lists of commitments included in the application with appropriate cross references. The industry understands that these lists will be used for site inspections and not for comparison with other applicants.

The industry has agreed to identify the high level future commitments in their (U)FSAR supplement (Appendix A of the LRA). Examples of what is meant by 'high level' can be seen in the enclosures. If at a later date (e.g., during an inspection) the NRC needs to review all commitments made during the license renewal process, they can obtain this information from the plant's commitment tracking system.

It is possible that applicants may differ as to how they will transmit this consolidated list to the NRC outside of the LRA (e.g., the programs that govern NRC correspondence at some plants may contain additional requirements such as identifying commitments specifically in a cover letter to the LRA). NEI feels that the way in which the information is transmitted to the NRC outside of the LRA,

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should remain a plant-specific preference and we are only recommending to our members that they identify their high level commitments in the (U) FSAR supplement section of the LRA. If plants decide to provide high level program descriptions (commitments) in the (U)FSAR supplement section of the LRA and identify other commitments in an attachment to the LRA cover letter that should be fine as well. PWR and BWR future commitments examples are provided in the enclosure.

If you have any questions, please call me (202) 739-8110 or by e-mail (apn@nei.org).

Sincerely,

Alan Nelson

Enclosures

Pressure Water Reactor
License Renewal Future Commitment Examples

- Develop and implement inspection program for buried piping and valves
- Add pressurizer surge line to Augmented Inspection Program
- Add core barrel hold-down spring to Augmented Inspection Program
- Expand scope of Civil Engineering Structural Inspection to cover License Renewal requirements
- Revise plant documents to use inspection opportunities when inaccessible areas become accessible during work activities
- Incorporate NFPA-25, Section 2-3.1.1 for sprinklers
- Develop inspection criteria for non-ASME supports and doors
- Develop procedural guidance for inspection criteria that puts focus on aging effects
- Develop and implement inspection program for infrequently accessed areas
- Develop and implement inspection program for tanks
- Follow industry activities related to failure mechanisms for small-bore piping. Evaluate changes to inspection activities based on industry recommendations
- Follow industry activities related to core support lugs. Evaluate need to enhance inspection activities based on industry recommendations
- Inspect representative sections of polar crane box girders
- Follow industry activities related to reactor vessel internals issues such as void swelling, thermal and neutron embrittlement, etc. Evaluate industry recommendations
- Implement changes into procedures to assure consistent inspection of components for aging effects during work activities
- Incorporate groundwater monitoring into the civil engineering structural monitoring program. Consider groundwater chemistry in engineering evaluations of deficiencies

Boiling Water Reactor
License Renewal Future Commitment Examples

- Evaluate any age related degradation found during recirculation system ISI inspections for applicability to the NSR portions of the recirculation system that was included in the scope of license renewal for NSR/SR.
- Notify the NRC whether Integrated Surveillance Program per BWRVIP-78 or plant specific program will be implemented
- Perform Inspection of carbon steel Component Supports (Other than ASME Class 1, 2, 3, and ASME Class MC component supports)
- Perform Inspection of SBO structural components
- Perform periodic reviews of calibration test results of electrical cables used in LPRM and WRM Instrumentation circuits to identify potential existence of aging degradation
- Perform inspection of outer sluice gates in the circulating water pump structure
- Perform inspection of hazard barrier doors in a sheltered environment for loss of material
- Perform inspection of RPV top guide
- Perform ultrasonic testing to detect wall thinning at susceptible locations in the ESW system stagnant piping in ECCS rooms
- Perform one-time inspection of a cast iron fire protection component for selective leaching
- Perform functional testing of sprinkler heads
- Perform inspection of electrical conduits in outdoor environment
- Perform inspection of Susquehanna substation wooden pole
- Perform one-time inspection of wall thickness of selected torus piping
- Perform inspection of PVC-insulated Fire Safe Shutdown cables in drywell
- Implement inspection program for Non-EQ accessible cables and connections, including fuse blocks

- Perform one-time piping inspection activities for standby liquid control system, auxiliary steam system, plant equipment and floor drain system, service water system, radiation monitoring system

- Perform one-time inspection of susceptible locations for loss of material in fuel pool cooling system to verify effectiveness of fuel pool chemistry activities

- Perform one-time inspection of carbon steel piping for loss of material in RPV instrumentation and Reactor Recirculation system

- Perform testing of inaccessible medium voltage cables

- Implement the final version of the fuse holder interim staff guidance when issued by the NRC.

- Implement fatigue management program

- Submit RPV P-T curves for 54 EFPY as license amendment

- Submit RPV circumferential weld examination relief request for 60 years

- Implement BWRVIP-76 when approved by the NRC and accepted by BWR VIP Committee

APPENDIX D

Standard
License Renewal Application
Format

Insert SLRA document

Appendix D

Standard License Renewal Application Format

May, 2003

Executive Summary

After receipt of the Calvert Cliffs and Oconee renewed Operating Licenses, both the NRC and the industry felt that the efficiency of the License Renewal Application (LRA) review process could be significantly improved. In early 2001, they began working together to develop a standard way of presenting the results of the aging management reviews in the LRA Section 3 tables; and the "Plant X and Plant Y" project was born.

Improvements in efficiency were realized through this effort. However, the industry and NRC felt that more still needed to be done in this area. NRC reviewers were experiencing some difficulty obtaining the required information in a format that could make their review most efficient. Numerous Requests for Additional Information (RAIs) were being issued for information that was already contained within the LRA, but which the reviewers were having difficulty locating. In addition, some information that the reviewers needed was not contained within the LRA. As each successive applicant attempted to address these needs, the presentation and content of data changed from application to application. This resulted in inefficiency and confusion for the reviewers and for the subsequent licensees who were developing applications.

In July of 2002, a group of utility members formed a Standard License Renewal Application project team, under the coordination of the Nuclear Energy Institute (NEI). The team met periodically with NRC staff throughout the remainder of 2002. Building upon the lessons learned from the "Plant X and Plant Y" effort, they developed a new set of Section 3 tables, which they believe contain the right amount of information, presented in the best way possible, in order to gain the maximum efficiency from the data presentation.

Since the various sections of the LRA work together to present the necessary information, it would not have been enough to only develop standard Section 3 tables. Therefore, the project team also standardized as much of the rest of the application as was necessary to gain the maximum efficiency for future NRC reviews. The result is a Standard License Renewal Application (SLRA) with examples of Section 2, Section 3, and Appendix B included. A "generic" Pressurized Water Reactor was used for all examples. This should not present a problem for Boiling Water Reactor applicants, since the purpose of the examples is to illustrate <u>format</u> and what <u>type</u> of information should be included in the LRA, not technical content.

The industry expects that all future applicants will use this guidance (as revised or incorporated into other industry documents) to develop their License Renewal Applications. The industry strongly urges NRC staff reviewers to honor this format and refrain from making special requests for format deviations.

Contents

This Appendix contains the following items:

Standard License
Renewal Application (SLRA)

Section 2.0

Section 2.0 Description

Introduction

The License Renewal Application (LRA) standardization project focused on those areas of the LRA where the industry and NRC felt that review efficiency could be significantly improved by standardizing the format. Section 2 was one of those areas. However, not all subsections within Section 2 needed attention. Where experience had demonstrated that this type of information is well known, no information was included in Section 2 of the Standard License Renewal Application (SLRA). For all other subsections of Section 2, sample information has been provided and presented in the format. The information contained within the SLRA is for illustrative purposes only. It is included to help the applicant and reviewer understand the subsection format and the type of information that is recommended for inclusion in a plant specific LRA. While the information may be technically valid for one plant, it may not be technically valid for another.

Format

Section 2 of the SLRA consists of the following four subsections:

- 2.0 Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review and Implementation Results
- 2.1 Scoping and Screening Methodology
- 2.2 Plant Level Scoping Results
- 2.3 Scoping and Screening Results

2.0 Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review and Implementation Results

This subsection provides a brief introduction to Section 2.0. In addition, it contains Table 2.0-1, "Intended Functions Abbreviations & Definitions," which contains the meanings for the abbreviations used in the Screening and AMR results tables to represent the intended functions for components, subcomponents and structural members.

2.1 Scoping and Screening Methodology

Experience has demonstrated that the expected format and content of this subsection are well understood by licensees and reviewers. Therefore, the SLRA contains no data for this section, with the exception of section 2.1.X.

During the SLRA development effort, the NRC informed the industry that their LRA review would be more efficient if the applicant would state their position regarding the subject of any Interim Staff Guidance (ISG) documents under development at the time of application submittal. Subsection 2.1.X has been included to reflect this recommendation. The numbering scheme of "2.1.X" indicates that this information should be contained somewhere within subsection 2.1, but exactly where in section 2.1 is at the discretion of the applicant.

2.2 Plant Level Scoping Results

Experience has demonstrated that the expected format and content of this subsection are well understood by licensees and reviewers. Therefore, the SLRA contains no data for this section.

2.3 Scoping and Screening Results

This section contains the scoping and screening results of the sample plant Containment Spray System, presented in the SLRA format. This subsection starts with number 2.3.2 because an Engineered Safety Features System subsystem was used (Containment Spray System) for the example. A Reactor Coolant System subsystem would be designated 2.3.1.

This section contains the following information for the sample plant Containment Spray System:

- System Description
- FSAR References
- License Renewal Drawings
- Components Subject to AMR

System Description

This section contains the Containment Spray System description. It incorporates NRC staff recommendations to include sufficient detail for the staff to use it in the associated section of the Safety Evaluation Report (SER). It includes a discussion of the system intended function (i.e., why the system is in scope for License Renewal), including which criteria of 10 CFR 54 require the system to be in scope. The portions of this system containing components subject to an AMR are also identified.

FSAR References

For the sample plant, only one section of the Final Safety Analysis Report (FSAR) is relevant to the scoping and screening for the Containment Spray System. However, if more sections of the FSAR were used, they would have been identified here. This section is hyperlinked to the appropriate FSAR section.

License Renewal Drawings

The license renewal drawings for the Containment Spray System are listed in this subsection. Each drawing number is hyperlinked to its associated drawing. Note that in the SLRA only the first two drawings have active hyperlinks.

Components Subject to AMR

The reviewer is referenced (with hyperlink) to Table 2.3.2-1 for a list of the component types that require Aging Management Review (AMR). In addition the reviewer is referenced to Table 3.2.2-1 (with hyperlink) for the results of the AMR of the Containment Spray System components.

2.3.2.X Plant Specific System

This section contains only formatting information. It is intended to illustrate where the next system within the Engineered Safety Features subsection would be located in a license renewal application.

Section 2.0 Example

2.0 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

This section provides the scoping and screening results for those component types that will be subject to aging management review in Section 3.0.

Definitions and abbreviations of the intended functions which were used in the scoping and screening and aging management reviews are included in Table 2.0-1, Intended Functions Abbreviations & Definitions.

INTENDED FUNCTIONS ABBREVIATIONS AND DEFINITIONS

This section contains the meanings for the abbreviations used in the Screening and AMR results tables to represent the intended functions for components, subcomponents, and structural members.

Table 2.0-1 Intended Functions Abbreviations & Definitions

Intended Function	Abbreviation	Definition
Conducts Electricity	CE	Conducts electricity.
Enclosure Protection	EN	Provides enclosure, shelter, or protection for in-scope equipment (including radiation shielding, pipe whip restraint, and thermal shielding).
EQ Barrier	EQB	Provides an environmental qualification (EQ) barrier.
Fire Barrier	FB	Provides a rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.
Flood Barrier	FLB	Provides a protective barrier for internal/external flood events.
Flow Control	FC	Provides flow control.

Table 2.0-1 Intended Functions Abbreviations & Definitions

Intended Function	Abbreviation	Definition
Flow Distribution	FD	Provides for flow distribution.
Flow Restriction	FR	Limits mass flowrate (water or steam). For example an orifice.
Filtration	FLT	Provides filtration.
Heat Sink	HS	Provides a heat sink during SBO or design basis accidents.
Heat Transfer	HT	Provides for heat transfer.
Jet Impingement Shield	JIS	Provides jet impingement shielding for high energy line breaks.
Missile Barrier	MB	Provides a missile (internal or external) barrier.
Pressure Boundary	PB	Provides a pressure boundary.
Source of Cooling Water	SCW	Provides a source of cooling water for plant shutdown.
Structural Support [Criteria a(2) & a(3) requirement]	SNS	Provides structural and/or functional support to equipment meeting license renewal Criterion 2 (non-safety affecting safety-related) and/or Criterion 3 (the five regulated events).
Spray Pattern	SP	Provides a spray pattern.
Structural Support [Criterion a(1) equipment]	SSR	Provides structural and/or functional support for safety-related equipment.

Table 2.0-1 Intended Functions Abbreviations & Definitions

Intended Function	Abbreviation	Definition
Vortex Suppression	VS	Suppressing or breaking the water vortex near pump suction to prevent cavitation.

2.1 SCOPING AND SCREENING METHODOLOGY

2.1.X INTERIM STAFF GUIDANCE DISCUSSION

The NRC staff has identified three issues for which additional staff and industry guidance clarification is necessary. They are:

1. Housing of Active Components
2. Interpretation of 10 CFR 54.4(a)(2)
3. Treatment of Electrical Fuse Holders

The following is a discussion of the general process used during the License Renewal Integrated Plant Assessment for each of these areas:

Housing of Active Components

The Statements of Consideration for 10 CFR 54 provides the License Renewal Rule philosophy that, during the extended period of operation, safety-related functions should be maintained in the same manner and to the same extent as during the current licensing term. Examples of structures and components that perform passive functions are listed in 10 CFR 54.21(a)(1)(ii), which states, "These structures and components include, but are not limited to, pump casings, valve bodies . . ."

Pumps and valves were just an example meant to focus the AMR process on the passive function of an SSC. That passive function is not limited to the pressure boundary of the reactor coolant system. The exclusion of an SSC due to its active nature only applies to that portion of the SSC with an active function and not to those portions of the SSC with a passive function. Therefore, an SSC such as a vent duct, which is both long-lived and passive, and houses a fire protection damper, is considered to be within the scope of license renewal and subject to aging management review.

Interpretation of 10 CFR 54.4(a)(2)

10 CFR 54.4(a)(2) states that SSCs within the scope of license renewal shall include non-safety related SSCs whose failure could prevent the satisfactory accomplishment of any of the functions identified for safety related SSCs, in 10 CFR 54.4(a)(1)(i)(ii) or (iii).

The process that was used to identify the in-scope non-safety related SSCs under 10 CFR 54.4(a)(2) was divided into two phases; an analytical phase and a physical phase. For the analytical phase, the FSAR, Technical Specifications, design documents, design drawings and the SSC safety classifications were reviewed to identify the in-scope non-safety related SSCs. However, this type of review alone could not provide the necessary information relative to system spatial interactions. Therefore, phase two utilized a plant spaces physical review, which evaluated SSCs for possible interactions that were not explicitly described in the CLB.

This two phased approach resulted in non-safety related SSCs being identified as in-scope when there was a potential for interaction either physically or spatially with the intended function of safety related SSCs.

Treatment of Electrical Fuse Holders

Consistent with the requirements specified in 10 CFR 54.4(a), fuse holders (including fuse clips and fuse blocks) are considered to be passive electrical components. Fuse holders are scoped, screened, and included in the aging management review (AMR) in the same manner as terminal blocks and other types of electrical connections. However, fuse holders inside the enclosure of an active component, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards, are considered to be piece parts of the larger assembly. Since piece parts and sub-components in such an enclosure are inspected regularly and maintained as part of the normal maintenance and surveillance activities, they are considered not subject to an AMR. Fuse holders perform a primary function similar to electrical connections by providing an electrical circuit to deliver rated voltage, current, or signals. These intended functions of fuses meet the criteria of 10 CFR 54.4(a). Additionally, these intended functions are performed without moving parts or without a change in configuration or properties as described in 10 CFR 54.21(a)(1)(i). Fuse holders are not replaced based on qualified life or specified time period, therefore fuse holders are passive, long-lived electrical components within the scope of license renewal and are subject to an AMR. Aging management of the clips is required for those cases where fuse holders are not considered piece parts of a larger assembly.

2.2 PLANT LEVEL SCOPING RESULTS

2.3 SCOPING AND SCREENING RESULTS: MECHANICAL SYSTEMS

2.3.2 ENGINEERED SAFETY FEATURES SYSTEMS

2.3.2.1 CONTAINMENT SPRAY (CS) SYSTEM

System Description

The purpose of the Containment Spray (CS) system is to limit the containment pressure and temperature after a Loss-of-Coolant Accident (LOCA) or Main Steam Line Break (MSLB) accident and thus reduce the possibility of leakage of airborne radioactivity to the outside environment. The CS system, in conjunction with the containment air recirculation and cooling system, provides sufficient heat removal capability to limit the post-accident containment pressure and structural temperature below the design values of 54 psig and 289°F, respectively.

The CS system initially draws borated treated water from the Refueling Water Storage Tank (RWST). The treated water flows through the CS pumps, shutdown cooling heat exchangers and interconnecting piping to the spray nozzles. When the RWST reaches a preset low level, the CS system draws off the containment sump through the CS pumps, shutdown cooling heat exchangers, and interconnecting piping to the spray nozzles. The CS nozzles direct sprays of cooled borated water downward from the upper regions of the containment to cool and depressurize the containment building.

The CS system meets 10 CFR 54.4 (a)(1) because it is a safety related system that is relied upon to remain functional during and following design-basis events, to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.67(b)(2) and 10 CFR 100.11. It also meets 10 CFR 54.4(a)(3) because the components within the system are relied upon in the safety analyses and plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for environmental qualification (10 CFR 50.49). The components subject to an AMR extend from the RWST and the Containment Sump system to the spray nozzles located inside containment.

FSAR References

Additional details of the CS system can be found in the FSAR, Section 6.4.

License Renewal Drawings

The license renewal drawings for the CS system are listed below:

25203-LR26015, Sh.1
25203-LR26015, Sh.2
25203-LR26015, Sh.3

Components Subject to AMR

The component types that require aging management review are indicated in Table 2.3.2-1, Containment Spray System.

The aging management review results for these components are provided in Table 3.2.2-1:, Engineered Safety Features - Containment Spray System - Summary of Aging Management Evaluation.

2.3.2.X PLANT SPECIFIC SYSTEM

System Description

FSAR References

Additional details of the **** System**** can be found in the FSAR, Section ****.

License Renewal Drawings

The license renewal drawings for the **** System **** are listed below:

Components Subject to AMR

The component types that require aging management review are indicated in Table 2.3.2-X , Plant Specific System.

Section 2 Tables: Engineered Safety Features

See Table 2.0-1 for definition of intended function

Table 2.3.2-1 Containment Spray System

Component Type	Intended Function(s)
Heat exchangers (shell)	Pressure boundary
Heat exchangers (tubes)	Heat transfer, Pressure boundary
Piping	Pressure boundary
Pump casing	Pressure boundary
Spray Nozzles	Flow control, Pressure boundary

See Table 2.0-1 for definition of intended function

Table 2.3.2-X Plant Specific System

Component Type	Intended Function(s)

See Table 2.0-1 for definition of intended function

Standard License Renewal Application (SLRA)

Section 3.0

Section 3.0 Description

Introduction

Section 3 is the area of License Renewal Application (LRA) that the industry and NRC felt needed the most standardization. Experience had shown that NRC reviewers were having difficulty obtaining the required information in a format conducive to an efficient review. Numerous Requests for Additional Information (RAIs) were being issued for information that was already in the LRA, but was not able to be located by the reviewer. In addition, some information that the reviewers needed was not in the LRA at all. As each successive applicant attempted to address these needs, the format and content changed from application to application. This resulted in inefficiency and confusion for the reviewers and for the subsequent licensees who were developing applications.

The LRA standardization effort focused on addressing these issues. The result is a Section 3 that includes "Boiler Plate" information in subsection 3.0 and a detailed example of a single subsection (Aging Management of Engineered Safety Features Systems) in subsection 3.2. It is expected that applicants will use the example subsection as a model for the remaining subsections of Section 3. The information contained within the Standard License Renewal Application (SLRA) is for illustrative purposes <u>only</u>. It is included to help the applicant and reviewer understand the SLRA subsection format and the <u>type</u> of information that is recommended for inclusion in a plant specific LRA. While the data may be technically valid for one plant, it may not be technically valid for another.

Format

Section 3 of the SLRA consists of the following two major subsections:

3.0 Aging Management Review Results
3.2 Aging Management of Engineered Safety Features Systems

The SLRA uses an Engineered Safety Features system (Containment Spray System) for its example. The SLRA project team felt that this would be the best example to use for the standardized LRA considering several factors, including availability of real plant data at the time of SLRA development. Therefore, the subsection numbering jumps from subsection 3.0 to 3.2, skipping subsection 3.1, which would be the Reactor Coolant System subsection. In fact, a licensee's LRA that is based on the SLRA is expected to actually contain six major subsections: 3.0 Aging Management Review Results, 3.1 Aging Management of Reactor Vessel, Internals, and Reactor Coolant System, 3.2 Aging Management of Engineered Safety Features, 3.3 Aging Management of Auxiliary Systems, 3.4 Aging Management of Steam and Power Conversion System, 3.5 Aging Management of Containment Structures and Component Supports, and 3.6 Aging Management of Electrical and Instrumentation and Controls.

3.0 Aging Management Review (AMR) Results

This subsection contains the roadmap for all of Section 3. It identifies where the tables are located (with hyperlinks) that identify the internal and external environments for the Systems, Structures and Components (SSCs) that were subject to aging management review. It also identifies where the table of definitions for abbreviations that were used in Section 3 is located (along with its hyperlink). In addition, it includes the following two subsections:

- Table Description
- Table Usage

Table Description

The purpose of Section 3 of the LRA is to present the results of the Aging Management Reviews. The Table Description section of the SLRA describes the two tables that have been developed to present the AMR results information. It describes each column and defines the type of information that each column should contain, including level of detail, where appropriate.

Table Usage

This section describes how the two tables work together to present all of the needed information to the reviewer.

3.2 Aging Management of Engineered Safety Features Systems

As was noted earlier, the SLRA uses an Engineered Safety Features system (Containment Spray System) for its example. It is expected that applicants will use this example as a model for the remaining subsections of Section 3.

Subsection 3.2 is further divided into the following subsections:

3.2.1 Introduction
3.2.2 Results
3.2.3 Conclusion
3.2.4 References

3.2.1 Introduction

This subsection provides the roadmap for the remainder of subsection 3.2. It lists the section of the SLRA where the Engineered Safety Features System SSCs are identified (including a hyperlink). It also lists the systems, or portions of systems, that are addressed in this subsection. Finally, it contains Table 3.2.1, which presents the sample plant Engineered Safety Features subsystem information, correlated to the data from NUREG-1801, Volume 1, "Table 2 Summary of Aging Management Programs for the

Engineered Safety Features Evaluated in Chapter V of the GALL Report."

3.2.2 Results

This subsection contains Table 3.2.2-1, which summarizes the results of the aging management reviews for the Containment Spray System. It also identifies where the same information would be located for the next system within the Engineered Safety Features System subsection (identified as "Plant Specific System"). There is no actual data contained within this subsection for the Plant Specific System. The Plant Specific System example is included for illustrative purposes only, so that the reader can determine how this subsection is to be constructed.

Subsection 3.2.2 also contains a summary of the materials, environments, aging effects requiring management and aging management programs for each subsystem within the Engineered Safety Features System.

Finally, it includes all of the "Further Evaluation Recommended" information associated with the Engineered Safety Features System. NUREG-1801 Volume 2 and the tables of the Standard Review Plan for License Renewal (NUREG-1800), indicate which attributes of the program need to be evaluated by the reviewer. This section provides the plant-specific information required for this evaluation.

3.2.3 Conclusion

This subsection contains a conclusion statement regarding the ability of the selected Aging Management Programs (AMPs) to manage the effects of aging on the SSCs that are subject to aging management review for the Engineered Safety Features System.

3.2.4 References

A list of references associated with Section 3.2 of the LRA is provided in this section.

Section 3.0 Example

3.0 AGING MANAGEMENT REVIEW RESULTS

This section provides the results of the aging management review for those structures and components identified in Section 2.0 as being subject to aging management review.

Descriptions of the internal and external service environments which were used in the aging management review to determine aging effects requiring management are included in Table 3.0-1, Internal Service Environments and Table 3.0-2, External Service Environments. The environments used in the aging management reviews are listed in the Environment column.

Most of the Aging Management Review (AMR) results information in Section 3 is presented in the following two tables:

- **Table 3.x.1** - where '3' indicates the LRA section number, 'x' indicates the subsection number from NUREG 1801, Volume 1, and '1' indicates that this is the first table type in Section 3. For example, in the Reactor Coolant System subsection, this table would be number 3.1.1, in the Engineered Safety Features subsection, this table would be 3.2.1, and so on. For ease of discussion, this table will hereafter be referred to in this Section as "Table 1."
- **Table 3.x.2-y** - where '3' indicates the LRA section number, 'x' indicates the subsection number from NUREG 1801, Volume 1, and '2' indicates that this is the second table type in Section 3; and 'y' indicates the system table number. For example, for the Reactor Vessel, within the Reactor Coolant System subsection, this table would be 3.1.2-1 and for the Reactor Vessel Internals, it would be table 3.1.2-2. For the Containment Spray System, within the Engineered Safety Features subsection, this table would be 3.2.2-1. For the next system within the ESF subsection, it would be table 3.2.2-2. For ease of discussion, this table will hereafter be referred to in this section as "Table 2."

<u>TABLE DESCRIPTION</u>

NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," contains the staff's generic evaluation of existing plant programs. It documents the technical basis for determining where existing programs are adequate without modification, and where existing programs should be augmented for the extended period of operation. The evaluation results documented in the report indicate that many of the existing programs are adequate to manage the aging effects for particular structures or components, within the scope of license renewal, without change. The report also contains recommendations on specific areas for which existing programs should be augmented for license renewal. In order to take full advantage of NUREG-1801, a comparison between the AMR results and the tables of NUREG-1801 has been made. The results of that comparison are provided in the two tables.

Table 1 (Figure 3.0-1)

The purpose of Table 1 is to provide a summary comparison of how the facility aligns with the corresponding tables of NUREG-1801, Volume 1. The table is essentially the same as Tables 1 through 6 provided in NUREG-1801, Volume 1, except that the "Type" column has been replaced by an "Item Number" column and the "Item Number in GALL" column has been replaced by a "Discussion" column.

The "Item Number" column provides the reviewer with a means to cross-reference from Table 2 to Table 1.

The "Discussion" column is used by the applicant to provide clarifying/amplifying information. The following are examples of information that might be contained within this column:

- "Further Evaluation Recommended" information or reference to where that information is located (including a hyperlink if possible)
- The name of a plant specific program being used (and a hyperlink to the program if possible)
- Exceptions to the NUREG-1801 assumptions
- A discussion of how the line is consistent with the corresponding line item in NUREG-1801, Volume 1, when that may not be intuitively obvious
- A discussion of how the item is different than the corresponding line item in NUREG-1801, Volume 1, when it may appear to be consistent (e.g., when there is exception taken to an aging management program that is listed in NUREG-1801, Volume 1)

The format of Table 1 provides the reviewer with a means of aligning a specific Table 1 row with the corresponding NUREG-1801, Volume 1 table row, thereby allowing for the ease of checking consistency.

Table 2 (Figure 3.0-2)

Table 2 provides the detailed results of the aging management reviews for those components identified in LRA Section 2 as being subject to aging management review. There will be a Table 2 for each of the subsystems within a "system" grouping. For example, for a PWR, the Engineered Safety Features System Group contains tables specific to Containment Spray, Containment Isolation, Emergency Core Cooling System, etc.

Table 2 consists of the following nine columns:

- Component Type
- Intended Function
- Material
- Environment
- Aging Effect Requiring Management
- Aging Management Programs
- NUREG-1801 Volume 2 Item
- Table 1 Item
- Notes

Component Type

The first column identifies all of the component types from Section 2 of the LRA that are subject to aging management review. They are listed in alphabetical order.

Intended Function

The second column contains the license renewal intended functions (including abbreviations where applicable) for the listed component types. Definitions and abbreviations of intended functions are contained within the Intended Functions table of LRA Section 2.

Material

The third column lists the particular materials of construction for the component type.

Environment

The fourth column lists the environment to which the component types are exposed. Internal and external service environments are indicated and a list of these environments is provided in the Internal Service Environments and External Service Environments tables of LRA Section 3.

Aging Effect Requiring Management

As part of the aging management review process, the applicant determines any aging effects requiring management for the material and environment combination in order to maintain the intended function of the component type. These aging effects requiring management are listed in column five.

Aging Management Programs

The aging management programs used to manage the aging effects requiring management are listed in column six of Table 2.

NUREG-1801 Vol. 2 Item

Each combination of component type, material, environment, aging effect requiring management, and aging management program that is listed in Table 2, is compared to NUREG-1801, Volume 2 with consideration given to the standard notes, to identify consistencies. When they are identified, they are documented by noting the appropriate NUREG-1801, Volume 2 item number in column seven of Table 2. If there is no corresponding item number in NUREG-1801, Volume 2, this row in column seven is left blank. That way, a reviewer can readily identify where there is correspondence between the plant specific tables and the NUREG-1801, Volume 2 tables.

Table 1 Item

Each combination of component, material, environment, aging effect requiring management, and aging management program that has an identified NUREG-1801 Volume 2 item number must also have a Table 3.x.1 line item reference number. The corresponding line item from Table 1 is listed in column eight of Table 2. If there is no corresponding item in NUREG-1801, Volume 1, this row in column eight is left blank. That way, the information from the two tables can be correlated.

Notes

In order to realize the full benefit of NUREG-1801, each applicant needs to identify how the information in Table 2 aligns with the information in NUREG-1801, Volume 2. This is accomplished through a series of notes. All note references with letters are standard notes that will be the same from application to application throughout the industry. Any notes the plant requires which are in addition to the standard notes will be identified by a number and deemed plant specific.

TABLE USAGE

Table 1

The reviewer evaluates each row in Table 1 by moving from left to right across the table. Since the Component, Aging Effect/Mechanism, Aging Management Programs and Further Evaluation Recommended information is taken directly from NUREG-1801, Volume 1, no further analysis of those columns is required. The information intended to help the reviewer the most in this table is contained within the Discussion column. Here the reviewer will be given information necessary to determine, in summary, how the applicant's evaluations and programs align with NUREG-1801, Volume 1. This may be in the form of descriptive information within the Discussion column or the reviewer may be referred to other locations within the LRA for further information (including hyperlinks where possible/practical).

Table 2

Table 2 contains all of the Aging Management Review information for the plant, whether or not it aligns with NUREG-1801. For a given row within the table, the reviewer is able to see the intended function, material, environment, aging effect requiring management and aging management program combination for a particular component type within a system. In addition, if there is a correlation between the combination in Table 2 and a combination in NUREG-1801, Volume 2, this will be identified by a referenced item number in column seven, NUREG-1801, Volume 2 Item. The reviewer can refer to the item number in NUREG-1801, Volume 2, if desired, to verify the correlation. If the column is blank, the applicant was unable to locate an appropriately corresponding combination in NUREG-1801, Volume 2. As the reviewer continues across the table from left to right, within a given row, the next column is labeled Table 1 Item. If there is a reference number in this column, the reviewer is able to use that reference number to locate the corresponding row in Table 1 and see how the aging management program for this particular combination aligns with NUREG-1801. There may be a hyperlink directly to the corresponding row in Table 1 as well.

Table 2 provides the reviewer with a means to navigate from the components subject to Aging Management Review (AMR) in LRA Section 2 all the way through the evaluation of the programs that will be used to manage the effects of aging of those components.

A listing of the abbreviations used in this section is provided in Section 1.4.1.

Table 3.0-1 Internal Service Environments

Environment	Description
Air	Dry/filtered compressed air (identified as Dry Air), non-dried compressed air, and atmospheric air (when internal to components such as ventilation system components, components open to atmosphere, etc.). Moisture-laden air conditions are noted, when applicable.
Gas	Nitrogen, oxygen, hydrogen, carbon dioxide, helium, freon, or Halon gases. Also includes vent gases from process systems.
Lubricating Oil	All lubricating oils used for in-scope plant equipment.
Fuel Oil	All fuel oils used for in-scope plant equipment.
Raw Water[1]	From a river, lake, pond, or groundwater source. Raw water is not demineralized or chemically treated to any significant extent. In general, raw water is rough filtered to remove large particles. Biocides may be added to raw water to control micro-organisms or macro-organisms. Other designations of raw water include water that leaks from any system and condensation.
Sea Water[1]	Water from a bay, sound, or ocean source. Sea water is not demineralized or chemically treated to any significant extent. In general, sea water is rough filtered to remove large particles. Biocides may be added to sea water to control micro-organisms or macro-organisms.

1. While these are considered internal environments for plant systems, they may also be identified as external environments for certain structural members and system components that are submerged.

Table 3.0-1 Internal Service Environments

Environment	Description
Treated water[1] (includes Steam)	Demineralized water or chemically purified water which is the source for water that may require further processing, such as for the primary or secondary coolant system. Treated water can be de-aerated, can include corrosion inhibitors, biocides, or boric acid, or can include a combination of treatments. Steam generated from treated water is included in this environment category. Examples of designations that are used to identify treated water in the Environment description sections of the aging management review results include: • treated water (borated water) - applies to primary systems water that is treated and monitored for quality under Primary Water Chemistry Aging Management Activity • treated water (component cooling) - applies to component cooling system water that is treated and monitored for quality under Closed-Cycle Cooling Water System Aging Management Activity • treated water (bearing cooling/chilled water) - applies to bearing cooling system and chilled water system water that is treated and monitored for quality under Closed-Cycle Cooling Water System Aging Management Activity • treated water (diesel cooling) - applies to local, self-contained diesel engine cooling water systems water that is treated and monitored for quality under Closed-Cycle Cooling Water System Aging Management Activity • treated water (secondary) - applies to secondary systems water that is within the scope of the Secondary Water Chemistry Aging Management Activity and controlled for protection of steam generators Other treated water applications use chemistry-controlled treated water as source water, but the water is not maintained as chemistry-controlled water.

1. While these are considered internal environments for plant systems, they may also be identified as external environments for certain structural members and system components that are submerged.

Table 3.0-2 External Service Environments

Environment[1]	Description
Air	Indoor air environments as described below: <u>Sheltered Air</u> - The sheltered air environment includes atmospheric air inside covered structures that provide protection from precipitation and wind. This environment is defined by a bulk average air temperature range of 40°F to 130°F and a 60-year maximum design ionizing dose of $3x10^7$ rads. <u>Containment Air</u> - The Containment air environment is defined by a bulk average air temperature range of 105°F to 120°F, except the pressurizer block house which can approach 150°F. Normal operating pressure is between -12 in. w.g. and 1.0 psig. The 60-year maximum design ionizing dose ranges between $6.6x10^5$ rads and $8.7x10^7$ rads. An exception is the area around the reactor vessel inside the primary shield wall for which the 60-year maximum design ionizing dose is $X.Xx10^9$ rads. <u>NOTES</u> 1.Certain structures or components may experience environmental conditions that deviate from the stated ranges or maximum values. The actual environmental condition(s) for these structures or components were used in the aging evaluation when the condition could affect the results, and, in those cases, the actual values are identified in the Environment description of the applicable LRA subsection. 2.Structural members may be associated with mechanical system components that may have the potential for condensation or intermittent wetting. Therefore, structural members have been conservatively assumed to be intermittently wetted in an air environment. 3.Mechanical components are assumed to be in an air environment that is not subject to intermittent wetting. Intermittently wetted conditions are noted, when applicable, such as from condensation.

1. For certain structural members and system components that are submerged, the applicable environment identified in Table 3.0-1, Internal Service Environments, is specified in the aging management review results.

Table 3.0-2 External Service Environments

Environment[1]	Description
Atmosphere / Weather	Air environment outside covered structures which includes precipitation and wind. Components and structures in this environment are subject to intermittent wetting. The outdoor air environment also includes exposure to ultraviolet radiation and ozone. This environment is bounded by a bulk average air temperature range of -5.1°F to 91°F and a 60-year maximum design ionizing dose of less than 150 rads.
Borated Water Leakage	The borated water leakage environment applies in all plant areas that include components and systems that contain borated water and that could leak on nearby components or structures. This environment is specified in the aging management review results only for materials susceptible to boric acid corrosion (carbon steel, low-alloy steels, and copper alloys). This environment is not considered for in-scope cables and connectors since cables are insulated, splices are sealed, and terminations are protected by enclosures.
Soil	The external environment for structures and components buried in the ground. Buried components (pipes and valves) are exposed to a soil environment and may be exposed to groundwater if they are located below the local groundwater elevation. The soil is assumed to entrain raw water and buried components are evaluated for the effects of corrosion. Concrete structural members below grade elevation are exposed to a soil environment and may be exposed to groundwater if they are located below the local groundwater elevation. The site groundwater is non-aggressive to concrete as determined by recent groundwater analyses. Steel piles are driven in undisturbed soil such that the soil environment surrounding the piles is deficient in oxygen at depths of a few feet below grade or below the water table. Therefore, the soil environment is not considered corrosive to steel piles.

1. *For certain structural members and system components that are submerged, the applicable environment identified in Table 3.0-1, Internal Service Environments, is specified in the aging management review results.*

Figure 3.0-1: Table 1

Table 3.x.1 Summary of Aging Management Evaluations in Chapter__ of NUREG-1801 for ___________

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.x.1- 01					
3.x.1- 02					
3.x.1- 03					
3.x.1- 04					
3.x.1- 05					
3.x.1- 06					

Figure 3.0-2: <u>Table 2</u>

Table 3.x.2- y   Section 3 Title - Plant Specific System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes

3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES SYSTEMS

3.2.1 INTRODUCTION

This section provides the results of the aging management review for those components identified in Section 2.3.2, Engineered Safety Features Systems, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

- Containment Spray (CS) System (Section 2.3.2.1)
- Plant Specific System (Section 2.3.2.X)

Table 3.2.1, Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features, provides the summary of the programs evaluated in NUREG-1801 for the Engineered Safety Features component groups that are relied on for license renewal.

This table uses the format described in Section 3.0 above. Note that this table only includes those component groups that are applicable to a PWR.

3.2.2 RESULTS

The following tables summarize the results of the aging management review for systems in the ESF system group.

Table 3.2.2-1:, Engineered Safety Features - Containment Spray System - Summary of Aging Management Evaluation

Table 3.2.2-X:, Engineered Safety Features - Plant Specific System - Summary of Aging Management Evaluation

The materials that components are fabricated from, the environments to which components are exposed, the potential aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.2.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.2.2.1.1, Containment Spray System

Section 3.2.2.1.X, Plant Specific System

3.2.2.1 MATERIALS, ENVIRONMENT, AGING EFFECTS REQUIRING MANAGEMENT AND AGING MANAGEMENT PROGRAMS

3.2.2.1.1 Containment Spray System

Materials

The materials of construction for the Containment Spray System components are:

- carbon steel
- stainless steel

Environment

The Containment Spray System components are exposed to the following environments:

- air
- borated water leakage
- nitrogen
- raw water
- treated water (borated)
- treated water

Aging Effects Requiring Management

The following aging effects, associated with the Containment Spray System, require management:

- loss of material
- fouling

Aging Management Programs

The following aging management programs manage the aging effects for the Containment Spray System components.

- *Boric acid corrosion*
- Heat exchanger monitoring
- System testing
- System walkdown
- Water chemistry control
- Oil analysis

3.2.2.1.X Plant Specific System

Materials

Environment

Aging Effects Requiring Management

Aging Management Programs

3.2.2.2 FURTHER EVALUATION OF AGING MANAGEMENT AS RECOMMENDED BY NUREG-1801

NUREG-1801 provides the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Engineered Safety Features, those programs are addressed in the following sections.

3.2.2.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3.

3.2.2.2.2.1 Loss of Material Due to General Corrosion

Applicable to BWR Only

3.2.2.2.2.2 Loss of Material Due to General Corrosion

Typically, general corrosion is managed the Chemistry Control Program for Primary Systems, supplemented by the Work Control Process. In certain instances, the Above Ground Carbon Steel Tanks inspections and Infrequently Accessed Area Inspection Activities also manage these aging mechanisms.

3.2.2.2.3.1 Local Loss of Material due to Pitting and Crevice Corrosion

Applicable to BWR Only

3.2.2.2.3.2 Local Loss of Material due to Pitting and Crevice Corrosion

In general, pitting and crevice corrosion is managed via the Chemistry Control Program for Primary Systems, supplemented by the Work Control Process. In certain instances, the Tank Inspection program inspections and Infrequently Accessed Area Inspection Activities also manage these aging mechanisms.

3.2.2.2.4 Local Loss of Material due to Microbiologically Influenced Corrosion

Microbiologically influenced corrosion (MIC) can occur in carbon steel or stainless steel that is in raw or treated water. Credit is given to the Chemistry Control Program for Primary Systems and Chemistry Control Program for Secondary Systems, supplemented by the Work Control Process, with the management of MIC in treated water systems. MIC is managed in the normally isolated Containment isolation valves and the piping in the service water system (raw water) with the Service Water System (Open-Cycle Cooling Water) inspections and the Work Control Process.

The Work Control Process provides the opportunity to visually inspect the internal surfaces of components and adjoining piping during preventive and corrective maintenance activities, and to correct any identified deficiencies promptly. The process supports various mitigation activities and allows for periodic sampling and trending of component conditions.

Therefore, the current aging management programs successfully manage MIC in Containment isolation valves and piping.

3.2.2.2.5 Changes in Properties due to Elastomer Degradation

Applicable to BWR Only

3.2.2.2.6 Local Loss of Material due to Erosion

In general, the Flow Accelerated Corrosion Program manages the loss of material due to erosion. In specific cases, the Fire Protection Program and the Service Water System (Open-Cycle Cooling Water) inspections manage this aging effect.

3.2.2.2.7 Buildup of Deposits due to Corrosion

Applicable to BWR Only

3.2.2.3 TIME-LIMITED AGING ANALYSIS

The time-limited aging analyses (TLAA) identified below are associated with the ESF systems components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Fatigue (Section 4.3, Metal Fatigue)
- Leak-before break (Section 4.7.3, Leak-Before-Break)

3.2.3 CONCLUSION

The ESF piping, fittings, and components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the ESF components are identified in the summary tables and Section 3.2.2.1.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the ESF components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

3.2.4 REFERENCES

1.

Results Tables: Engineered Safety Features

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1- 01	Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	This TLAA is further evaluated in Section 4.3. Low temperature portions are not susceptible to cumulative fatigue damage, for example, core flood. Further evaluation documented in Subsection 3.2.2.2.1
3.2.1- 02	BWR Only				
3.2.1- 03	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801 for containment isolation. System walkdown program is credited. For further evaluation, see Appendix B. Not applicable for containment spray and ECCS as these components are not carbon steel in these systems. Further evaluation documented in Subsection 3.2.2.2.2.2
3.2.1- 04	BWR Only				

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1- 05	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801 for containment isolation. Containment leak rate, wall thinning, and system walkdown programs are credited. For further evaluation, see Appendix B. Not applicable for containment spray and ECCS as these components are not carbon steel in these systems. Further evaluation documented in Subsection 3.2.2.2.3.2
3.2.1- 06	Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion (MIC)	Plant specific	Yes, plant specific	Consistent with NUREG-1801. Water chemistry control program is credited. For further evaluation, see Appendix B. Further evaluation documented in Subsection 3.2.2.2.4
3.2.1- 07	BWR Only				

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1- 08	High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific	Not applicable as HPSI and LPSI pumps are not normally in use. Further evaluation documented in Subsection 3.2.2.2.6
3.2.1- 09	BWR Only				
3.2.1- 10	External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Consistent with NUREG-1801. System walkdown program is credited. See Appendix B. Further evaluation documented in Subsection 3.2.2.2.2.2
3.2.1- 11	Piping and fittings of CASS in emergency core cooling systems	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	Not applicable as CASS is not used in this system.

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1- 12	Components serviced by open-cycle cooling system	Local loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to *biofouling*	Open-cycle cooling water system	No	Different programs are credited other than an open-cycle cooling water system. These are the heat exchanger monitoring, water chemistry control, and/or system testing programs. See Appendix B.
3.2.1-13	Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	Different programs are credited other than a closed-cycle cooling water system. These are water chemistry control, heat exchanger monitoring, and/or metal fatigue TLAA. See Appendix B.
3.2.1- 14	BWR Only				

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1-15	Pumps, valves, piping, and fittings, and tanks in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No	Consistent with NUREG-1801 where applicable. Not applicable for systems where temperature is below threshold for cracking.
3.2.1-16	BWR Only				
3.2.1-17	Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	Boric acid corrosion program has exceptions to NUREG-1801 AMP. See Appendix B.
3.2.1- 18	Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	This item is not applicable as temperatures in these systems are not high enough to cause these aging effects.

Table 3.2.2-1: Engineered Safety Features - Containment Spray System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Heat exchangers (shell)	PB	Carbon steel	Air (external)	Loss of material	System walkdown	V.E.1-b	3.2.1- 10	A
			Borated water leakage (external)	Loss of material	Boric acid corrosion	V.A.6-d	3.2.1-17	B
			Raw water (internal)	Loss of material	Heat exchanger monitoring Water chemistry control	V.A.6-a	3.2.1- 12	E
			Treated water (internal)	Loss of material	Water chemistry control	V.A.6-c	3.2.1-13	E

See Table 2.0-1 for definitions of intended function, Table 3.0-1 for definitions of internal environments and Table 3.0-2 for definitions of external environments.

Table 3.2.2-1: Engineered Safety Features - Containment Spray System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Heat exchangers (tubes)	HT	Stainless steel	Raw water (external)	Fouling	System testing Water chemistry control	V.A.6-b	3.2.1- 12	E, 2
			Treated water (external)	Fouling	Water chemistry control			H, 2
			Treated water (borated) (internal)	Fouling	Water chemistry control			H, 2
	PB	Stainless steel	Raw water (external)	Loss of material	Water chemistry control	V.A.6-a	3.2.1- 12	E
				Loss of material	Heat exchanger monitoring			H
			Treated water (external)	Loss of material	Heat exchanger monitoring	V.A.6-c	3.2.1-13	E
				Loss of material	Water chemistry control			H
			Treated water (borated) (internal)	Loss of material	Water chemistry control	V.A.6-a V.A.6-c	3.2.1- 12 3.2.1-13	E E, 3

See Table 2.0-1 for definitions of intended function, Table 3.0-1 for definitions of internal environments and Table 3.0-2 for definitions of external environments.

Table 3.2.2-1: Engineered Safety Features - Containment Spray System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping	PB	Stainless steel	Air (external)	None	None			G
			Air (internal)	None	None			G
			Nitrogen (internal)	None	None			G
			Treated water (borated) (internal)	Loss of material	Water chemistry control			H, I, 1
Pump casing	PB	Stainless steel	Air (external)	None	None			G
			Treated water (borated) (internal)	Loss of material	Water chemistry control			H, I, 1
Spray nozzles	FC PB	Stainless steel	Air (external and internal)	None	None			G

See Table 2.0-1 for definitions of intended function, Table 3.0-1 for definitions of internal environments and Table 3.0-2 for definitions of external environments.

Table 3.2.2-X: Engineered Safety Features - Plant Specific System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes

See Table 2.0-1 for definitions of intended function, Table 3.0-1 for definitions of internal environments and Table 3.0-2 for definitions of external environments.

Notes for Tables 3.2.2-1 through 3.2.2-X

A. Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

B. Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C. Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

D. Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

E. Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited.

F. Material not in NUREG-1801 for this component.

G. Environment not in NUREG-1801 for this component and material.

H. Aging effect not in NUREG-1801 for this component, material and environment combination.

I. Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J. Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

1. The system temperature is below the threshold for cracking.
2. Fouling is not restricted to biofouling only.
3. NUREG-1801 differentiates between open and closed systems; however, both have borated water internally.
4. Component type, material, environment and aging effect combination not in NUREG-1801, but aging management program in NUREG-1801 is used.

See Table 2.0-1 for definitions of intended function, Table 3.0-1 for definitions of internal environments and Table 3.0-2 for definitions of external environments.

Standard License Renewal Application (SLRA)

Appendix B

Appendix B Description

Introduction

The License Renewal Application (LRA) standardization project focused on those areas of the LRA where the industry and NRC felt that review efficiency could be significantly improved by standardizing the format. Appendix B was one of those areas. The information contained within the Standard License Renewal Application (SLRA) is for illustrative purposes only. It is included to help the applicant and reviewer understand the subsection format and the type of information that is recommended for inclusion in a plant specific LRA. While the data may be technically valid for one plant, it may not be technically valid for another.

Format

Appendix B of the SLRA consists of the following four subsections:

B1.0 Introduction
B2.0 Aging Management Programs
B3.0 TLAA Evaluation of Aging Management Programs Under 10 CFR 54.21(c)(1)(iii)
B4.0 References

B1.0 Introduction

This section provides an overview of Appendix B and provides general information to be used by the reviewer while navigating through Appendix B. It contains the following subsections:

- B1.1 Overview
- B1.2 Method of Discussion
- B1.3 Quality Assurance and Administrative Controls
- B1.4 Operating Experience
- B1.5 Aging Management Programs

B1.1 Overview

This subsection provides a general overview of the Introduction section of Appendix B.

B1.2 Method of Discussion

This subsection addresses the method for describing the Aging Management Programs (AMPs) in Appendix B. As part of the SLRA development effort, it was decided that there would be three ways to classify an aging management program. It would be either "consistent with NUREG-1801," "consistent with NUREG-1801, with exceptions" or "plant specific." This section states how the program descriptions will differ between those programs that are consistent with NUREG-1801 or are consistent with exceptions, and those that are plant specific.

B1.3 Quality Assurance Program and Administrative Controls

This subsection describes how the Quality Assurance Program and the plant Administrative Controls support License Renewal. It includes a discussion of the following topics:

- Corrective Actions
- Confirmation Process
- Administrative Controls

These items are discussed in the Introduction section of Appendix B because they apply throughout Appendix B.

B1.4 Operating Experience

This subsection describes how operating experience (industry and plant specific) was incorporated into the License Renewal process. This discussion is included in the Introduction section of Appendix B, because it applies throughout Appendix B.

B1.5 Aging Management Programs

This subsection contains a list of the sample plant programs credited for License Renewal. The programs are listed in alphabetical order and include a hyperlink to the section of Appendix B that contains the program description.

Only three programs are actually described in the SLRA. They are the "ASME Section XI, Subsections IWB, IWC & IWD Inservice Inspection Program" (consistent with NUREG-1801, with exceptions), the "Periodic Surveillance and Preventive Maintenance Program" (plant specific program) and the "Environmental Qualification Program" (consistent with NUREG 1801). All other programs in the list illustrate that they would be hyperlinked via highlighted text, but the hyperlinks do not actually function in the SLRA.

For each program in the list, there is an indicator in brackets that identifies the program as either existing or as being a new program for License Renewal.

B2.0 Aging Management Programs

This section contains a table that identifies the sample plant aging management programs, along with the corresponding NUREG-1801 program number and name (if applicable). The programs are listed in the program order of NUREG-1801. The programs that are consistent with NUREG-1801 or are consistent with exceptions, are listed first; followed by the plant specific programs. This section contains the following subsections:

- B2.1.1 ASME Section XI, Subsections IWB, IWC & IWD Inservice Inspection Program
- B2.1.2 ASME Section XI, Subsections IWE & IWL Inservice Inspection Program

B2.1.1 ASME Section XI, Subsections IWB, IWC & IWD Inservice Inspection Program

This program is consistent with exceptions to a NUREG-1801 program. Therefore, this section contains the following subsections:

- Program Description
- NUREG-1801 Consistency
- Exceptions to NUREG-1801
- Enhancements
- Operating Experience
- Conclusion

Program Description

This subsection contains a general description of the Aging Management Program (AMP).

NUREG-1801 Consistency

This subsection contains a statement regarding how consistent the plant specific AMP is with the corresponding NUREG-1801 AMP.

Exceptions to NUREG-1801

This subsection identifies the elements of the plant specific AMP that are not consistent with the NUREG-1801 AMP description. In addition, it describes why those exceptions are needed.

Enhancements

This subsection identifies enhancements that will be made to the plant specific program, processes or procedures in order to manage the effects of aging on the associated SSCs during the period of extended operation. Specific program elements that will be enhanced are identified.

Operating Experience

This subsection discusses operating experience related to the program.

Conclusion

This subsection states a general conclusion as to how the program will manage the effects of aging on the SSCs within the scope of the program, for the period of extended operation.

B2.1.1 Periodic Surveillance and Preventive Maintenance Program

This is a plant specific AMP. Therefore, this section contains the following subsections:

- Program Description
- Aging Management Program Elements
- Enhancements
- Conclusion

Program Description

This subsection contains a general description of the plant specific AMP.

Aging Management Program Elements

This subsection contains a discussion of the plant specific AMP in terms of the ten program elements of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants."

Enhancements

This subsection identifies any enhancements that will be made to the plant specific program, processes or procedures in order to manage the effects of aging on SSCs during the period of extended operation. Specific program elements that will be enhanced are identified.

Conclusion

This subsection states a general conclusion as to how the program will manage the effects of aging on the associated SSCs, for the period of extended operation.

B3.0 TLAA Evaluation of Aging Management Programs Under 10 CFR 54.21(c)(1)(iii)

This section addresses programs credited in the evaluation of Time Limited Aging Analyses (TLAAs). Only one example program is included in this section of the SLRA (B3.1 Environmental Qualification Program). It is provided as a model for any additional TLAA programs that would be included in the LRA.

This section contains the following subsection:

- B3.1 Environmental Qualification Program

If applicants have additional TLAA AMPs, those programs would be addressed sequentially as B3.2, B3.3, B3.4, etc.

B3.1 Environmental Qualification Program

This section describes the sample plant Environmental Qualification Program as it pertains to managing the effects of aging on the associated in-scope SSCs, for the period of extended operation. Since it is consistent with a NUREG-1801 AMP, this section is divided into the following subsections:

- Program Description
- NUREG-1801 Consistency
- Exceptions to NUREG-1801
- Enhancements
- Operating Experience
- Conclusion

B4.0 References

This section lists all of the references used throughout Appendix B.

Appendix B Example

B1.0 INTRODUCTION

B1.1 OVERVIEW

License Renewal aging management program descriptions are provided in this appendix for each program credited for managing aging effects based upon the aging management review results provided in Sections 3.1 through 3.6 of this application.

Each aging management program described in this section has ten elements which are consistent with the definitions in Section A.1, "Aging Management Review - Generic," Table A.1-1, "Elements of an Aging Management Program for License Renewal," of the NUREG-1800, SRP-LR (Reference 1). The 10 element detail is only provided when the program is plant specific. See [Section B1.2] below.

B1.2 METHOD OF DISCUSSION

For those Aging Management programs that are consistent with the assumptions made in Sections X and XI of NUREG-1801, or are consistent with exceptions, each program discussion is presented in the following format:

- A Program Description abstract of the overall program form and function is provided.
- A NUREG-1801 Consistency statement is made about the program.
- *Exceptions to the NUREG-1801 program are outlined and a justification is provided.*
- Enhancements to ensure consistency with NUREG-1801 <u>or</u> additions to the NUREG-1801 program to manage aging for additional components with aging effects not assumed in NUREG-1801 for the NUREG-1801 program. A proposed schedule for completion is discussed.
- Operating Experience information specific to the program is provided.
- A Conclusion section provides a statement of reasonable assurance that the program is effective, or will be effective, once enhanced.

For those programs that are plant specific, the above form is generally followed with the additional discussion of each of the ten elements.

B1.3 QUALITY ASSURANCE PROGRAM AND ADMINISTRATIVE CONTROLS

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800 (Reference 1). The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related systems, structures, and components that are subject to aging management review. In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. Generically the three elements are applicable as follows:

Corrective Actions:

A single corrective actions process is applied regardless of the safety classification of the structure or component. Corrective actions are implemented through the initiation of an Action Request (AR) in accordance with plant procedures established in response to 10 CFR 50, Appendix B. Plant procedures require the initiation of an AR for actual or potential problems, including unexpected plant equipment degradation, damage, failure, malfunction or loss. Site documents that implement aging management activities for license renewal will direct that an AR be prepared in accordance with those procedures whenever non-conforming conditions are found (i.e., the acceptance criteria are not met).

Equipment deficiencies are corrected through the initiation of a Work Order (WO) in accordance with plant procedures. Although equipment deficiencies may initially be documented by a WO, the corrective action process specifies that an AR also be initiated if required.

Confirmation Process:

The focus of the confirmation process is on the follow-up actions that must be taken to verify effective implementation of corrective actions. The measure of effectiveness is in terms of correcting the adverse condition and precluding repetition of significant conditions adverse to quality. Plant procedures include provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause determinations and prevention of recurrence where appropriate (e.g., significant conditions adverse to quality). These procedures provide for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken. The AR process is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of an AR. The aging management activities required for license renewal would also uncover any unsatisfactory condition due to ineffective corrective action.

Since the same 10 CFR 50, Appendix B corrective actions and confirmation process is applied for nonconforming SR and NSR structures and components subject to an Aging Management Review (AMR) for license renewal, the corrective action program is consistent with the NUREG-1801 elements.

Administrative Controls:

Administrative controls procedures provide information on procedures and other forms of administrative control documents, as well as guidance on classifying documents into the proper document type. Procedure attachments provide a chart showing the administrative controls hierarchy and a document type decision tree.

B1.4 OPERATING EXPERIENCE

Industry operating experience was incorporated into the License Renewal process through a review of industry documents to identify aging effects and mechanisms that could challenge the intended function of systems and structures within the scope of License Renewal. Review of plant specific operating experience was performed to identify aging effects experienced. The review of plant specific operating experience involved electronic database searches of plant information. In addition, discussions with system engineers and long time company employees were conducted, and identified some additional aging concerns.

Operating experience of the program/activity, including past corrective actions resulting in program enhancements, were considered. This information provides objective evidence that the effects of aging have been, and will continue to be, adequately managed.

B1.5 AGING MANAGEMENT PROGRAMS

The following aging management programs are described in the sections listed in this appendix. The programs are either discussed in NUREG -1801 or are site specific. Plant specific programs are listed at the end of the table in Section B2.0. Programs are identified as either existing or new.

1. ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program [Section B2.1.1] [Existing]
2. ASME Section XI, Subsections IWE & IWL Inservice Inspection Program [Section B2.1.2] [Existing]
3. ASME Section XI, Subsection IWF Inservice Inspection Program [Section B2.1.3] [Existing]
4. Bolting Integrity Program [Section B2.1.4] [New]
5. Boraflex Monitoring Program [Section B2.1.5] [New]
6. Boric Acid Corrosion Program [Section B2.1.6] [Existing]
7. Buried Services Monitoring Program [Section B2.1.7] [New]
8. Cable Condition Monitoring Program [Section B2.1.8] [New]
9. Closed-Cycle Cooling Water System Surveillance Program [Section B2.1.9] [New]

10. Fire Protection Program [Section B2.1.11] [Existing]

11. Flow-Accelerated Corrosion Program [Section B2.1.12] [Existing]

12. Fuel Oil Chemistry Control Program [Section B2.1.13] [Existing]

13. One-Time Inspection Program [Section B2.1.14] [New]

14. Open-Cycle Cooling (Service) Water System Surveillance Program [Section B2.1.15][New]

15. Periodic Surveillance and Preventive Maintenance Program [Section B2.1.2] [Existing]

16. Reactor Coolant System CASS Embrittlement Program [Section B2.1.17] [New]

17. Reactor Coolant System Alloy 600 Inspection Program [Section B2.1.18] [New]

18. Reactor Vessel Internals Program [Section B2.1.19] [Enhanced]

19. Reactor Vessel Surveillance Program [Section B2.1.20] [Existing]

20. Steam Generator Integrity Program [Section B2.1.21] [Existing]

21. Structures Monitoring Program [Section B2.1.22] [Existing]

22. Systems Monitoring Program [Section B2.1.23] [Existing]

23. Tank Internal Inspection Program [Section B2.1.24] [New]

24. Thimble Tube Inspection Program [Section B2.1.25] [Existing]

25. Water Chemistry Control Program [Section B2.1.26] [Existing]

B1.6 TIME LIMITED AGING ANALYSES AGING MANAGEMENT PROGRAMS:

1. Environmental Qualification Program [Section B3.1] [Existing]

2. Fatigue Monitoring Program [Section B3.2] [Existing]

3. Pre-Stressed Concrete Containment Tendon Surveillance Program [Section B3.3] [Existing]

B2.0 AGING MANAGEMENT PROGRAMS

The correlation between NUREG-1801 (Generic Aging Lessons Learned (GALL) programs and sample plant programs is shown below. For the sample plant programs, links to appropriate sections of this appendix are provided.

NUREG-1801 NUMBER	NUREG-1801 PROGRAM	PLANT PROGRAM
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, & IWD	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program [Section B2.1.1]
XI.M2	Water Chemistry	Water Chemistry Control Program [Section B2.1.26]
XI.M3	Reactor Head Closure Studs	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program [Section B2.1.1]
XI.M4	BWR Vessel ID Attachment Welds	Not Applicable, Sample is a PWR.
XI.M5	BWR Feedwater Nozzle	Not Applicable, Sample is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not Applicable, Sample is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not Applicable, Sample is a PWR.
XI.M8	BWR Penetrations	Not Applicable, Sample is a PWR.
XI.M9	BWR Vessel Internals	Not Applicable, Sample is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Program [Section B2.1.6]
XI.M11	Nickel-Alloy Nozzles and Penetrations	Reactor Coolant System Alloy 600 Inspection Program [Section B2.1.18]
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Reactor Coolant System CASS Embrittlement Program [Section B2.1.17]
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Reactor Vessel Internals Program [Section B2.1.19]

NUREG-1801 NUMBER	NUREG-1801 PROGRAM	PLANT PROGRAM
XI.M14	Loose Parts Monitoring	Not Applicable-not credited for aging management. Reactor Vessel Internals Program [Section B2.1.19] was determined to be adequate to manage identified aging effects.
XI.M15	Neutron Noise Monitoring	Not Applicable-not credited for aging management. Reactor Vessel Internals Program [Section B2.1.19] was determined to be adequate to manage identified aging effects.
XI.M16	PWR Vessel Internals	Reactor Vessel Internals Program [Section B2.1.19]
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion Program [Section B2.1.12]
XI.M18	Bolting Integrity	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program [Section B2.1.1] Systems Monitoring Program [Section B2.1.23] Structures Monitoring Program [Section B2.1.22] ASME Section XI, Subsection IWF Inservice Inspection Program [Section B2.1.3] Periodic Surveillance and Preventive Maintenance Program [Section B2.1.2]
XI.M19	Steam Generator Tube Integrity	Steam Generator Integrity Program [Section B2.1.21]
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling (Service) Water System Surveillance Program [Section B2.1.15]
XI.M21	Closed-Cycle Cooling Water System	Closed-Cycle Cooling Water System

NUREG-1801 NUMBER	NUREG-1801 PROGRAM	PLANT PROGRAM
		Surveillance Program [Section B2.1.9]
XI.M22	Boraflex Monitoring	Boraflex Monitoring Program [Section B2.1.5]
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Structures Monitoring Program [Section B2.1.22]
XI.M24	Compressed Air Monitoring	Not Credited for Aging Management
XI.M25	BWR Reactor Water Cleanup System	Not Applicable, Sample is a PWR.
XI.M26	Fire Protection	Fire Protection Program [Section B2.1.11]
XI.M27	Fire Water System	Fire Protection Program [Section B2.1.11]
XI.M28	Buried Piping and Tanks Surveillance	Not Applicable-See XI.M34.
XI.M29	Aboveground Carbon Steel Tanks	Systems Monitoring Program [Section B2.1.23] Tank Internal Inspection Program [Section B2.1.24]
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry Control Program [Section B2.1.13]
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance Program [Section B2.1.20]
XI.M32	One-Time Inspection	One-Time Inspection Program [Section B2.1.14]
XI.M33	Selective Leaching of Materials	One-Time Inspection Program [Section B2.1.14]
XI.M34	Buried Piping and Tanks Inspection	Buried Services Monitoring Program [Section B2.1.7]
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Cable Condition Monitoring Program [Section B2.1.8]

NUREG-1801 NUMBER	NUREG-1801 PROGRAM	PLANT PROGRAM
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Cable Condition Monitoring Program. [Section B2.1.8]
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Cable Condition Monitoring Program [Section B2.1.8]
XI.S1	ASME Section XI, Subsection IWE	ASME Section XI, Subsections IWE & IWL Inservice Inspection Program [Section B2.1.2]
XI.S2	ASME Section XI, Subsection IWL	ASME Section XI, Subsections IWE & IWL Inservice Inspection Program[Section B2.1.2]
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF Inservice Inspection Program [Section B2.1.3]
XI.S4	10 CFR 50, Appendix J	ASME Section XI, Subsections IWE & IWL Inservice Inspection Program [Section B2.1.2]
XI.S5	Masonry Wall Program	Structures Monitoring Program [Section B2.1.22]
XI.S6	Structures Monitoring Program	Structures Monitoring Program [Section B2.1.22]
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Structures Monitoring Program [Section B2.1.22]
XI.S8	Protective Coating Monitoring and Maintenance	Not Applicable-no credit is taken for protective coatings inside containment to prevent aging effects.
Chapter X		
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring Program [Section B3.2]

NUREG-1801 NUMBER	NUREG-1801 PROGRAM	PLANT PROGRAM
X.E1	Environmental Qualification (EQ) of Electrical Components	Environmental Qualification Program [Section B3.3]
X.S1	Concrete Containment Tendon Prestress	Pre-Stressed Concrete Containment Tendon Surveillance Program [Section B3.1]
NA	Plant Specific Program	Thimble Tube Inspection Program [Section B2.1.25]
NA	Plant Specific Program	Tank Internal Inspection Program [Section B2.1.24]
NA	Plant Specific Program	Periodic Surveillance and Preventive Maintenance Program [Section B2.1.2]
NA	Plant Specific Program	Systems Monitoring Program [Section B2.1.23]

B2.1 AGING MANAGEMENT PROGRAMS

B2.1.1 ASME SECTION XI, SUBSECTIONS IWB, IWC, AND IWD INSERVICE INSPECTION PROGRAM

Program Description

The ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection (ISI) Program inspections are performed to identify and correct degradation in Class 1, 2, and 3 piping, components and their integral attachments. The program includes periodic visual, surface and/or volumetric examinations and leakage tests of all Class 1, 2 and 3 pressure-retaining components, and their integral attachments, including welds, pump casings, valve bodies, and pressure-retaining bolting. These components and their integral attachments are identified in ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, or commitments requiring augmented inservice inspections in accordance with ASME Section XI, and are within the scope of License Renewal.

NUREG-1801 Consistency

The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program is an existing program that is consistent with exceptions, to NUREG-1801, Section XI.M1 (ASME Section XI Inservice Inspections Program, Subsections IWB, IWC, and IWD), and XI.M3 (Reactor Head Closure Studs) (Reference 3).

Exceptions to NUREG-1801

The exception to NUREG-1801 is that the sample plant ISI Program is based on the 1998 edition through 2000 addenda of ASME Section XI. NUREG-1801 uses the 1989 edition of ASME Section XI with 1995 edition through 1996 addenda changes noted. Use of the later edition code was found acceptable by the NRC in an SER dated November 5, 2001 (Reference 4).

<u>Program Elements Affected</u>

- **Detection of Aging Effects**
  The NDE techniques used to inspect the Class 1 (Table IWB-2500-1 components), Class 2 (Table IWC-2500-1 components), and Class 3 (Table IWD-2500-1 components) are consistent with the referenced ASME Section XI Code for those components. Therefore, the inservice inspections performed are consistent with the NUREG-1801 program except for the differences listed below.

  The sample plant ISI Program for the 4th Inspection Interval will meet the requirements of the 1998 edition through 2000 addenda of ASME Section XI, as modified by Risk Informed Inservice Inspection (RI-ISI) criteria for Examination Categories B-J, B-F, C-F-1, and C-F-2, and the additional requirements of 10 CFR 50.55a. The periodicity of most examinations is once per interval, with the exception of examination categories B-P, C-H, and D-B (98A00 edition). These include visual VT-2 examination of all pressure-retaining components each period during the system leakage test. The sample plant ISI Program specifies performance of the system leakage test once per refueling outage or each period for these examination categories, whichever applies.

a. Class 1, 2 and 3 Butt-Welded Piping

For Class 1, 2 and 3 butt-welded piping, the sample plant will implement RI-ISI. RI-ISI can reduce the number of piping welds where volumetric examination is required, based on an assessment of the probability of failure of these welds and the safety consequences of failure of these welds. RI-ISI can completely eliminate the surface examination of piping butt welds from the ISI Program. RI-ISI will credit enhancements to the VT-2 examinations performed during each refueling outage as well as enhancements to routine visual examinations performed by operations and system engineering personnel. RI-ISI applies to Categories B-J, B-F, C-F-1 and C-F-2 (98A00 edition). (Note: It will not apply to Class 3 components.)

b. Reactor Vessel Head Closure Studs

The sample plant's treatment of Reactor Vessel Head Closure Studs is consistent with ASME Section XI 1998 edition through 2000 addenda, which requires either a surface or volumetric examination, but not both. Volumetric examinations are performed using Performance Demonstration Initiative (PDI) techniques in accordance with ASME Section XI, Appendix VIII and 10 CFR 50.55a.

c. Class 1, 2 and 3 Pressure-Retaining Bolting

This element is consistent with the NUREG-1801 AMP for Class 1, 2 and 3 pressure-retaining bolting, except that Risk-Informed ISI will be implemented for Examination Categories B-F, B-J, C-F-1, and C-F-2, for the 4th Inspection Interval. The version of ASME Section XI referenced in NUREG-1801 AMP XI.M3, requires volumetric and surface exams of Reactor Vessel Head Closure Studs. The 98A00 version of the code only requires surface or volumetric examinations, but not both. Also, elimination of the surface examination will reduce the potential for handling-induced defects to the studs.

Enhancements

Enhancements to the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program include revisions to existing activities which will be credited for license renewal to ensure that the applicable aging effects are discovered and evaluated.

Program Elements Affected

Revise applicable existing procedures to ensure that the procedures address the following elements:

- **Corrective Actions**
  Documents that implement aging management activities for license renewal shall direct that an Action Request (AR) be prepared in accordance with plant procedures whenever the acceptance criteria are not met.

Enhancements are scheduled for completion prior to the period of extended operation.

Operating Experience

Both industry and sample plant-specific operating experience relating to the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program was reviewed.

A search of action requests and Maintenance Work Orders on Reactor Vessel Head Closure Studs for both Units revealed that no degradation of the studs or nuts was present. The examinations and inspections are conducted according to the requirements specified in Table IWB-2500-1.

The review of plant-specific operating experience revealed two instances where ISI examinations discovered flaws through means other than the system leakage test. Flaw indications were discovered in each reactor vessel outlet nozzle-to-shell weld during the ultrasonic examination of reactor vessel welds. A fracture mechanics evaluation was performed that demonstrated that the flaws posed no threat to continued safe operation of the reactor vessel.

As a result of industry experience review and additional examinations, some degradation (a crack) was discovered via radiography performed on a Masoneilan containment isolation valve seat cavity. The radiographic examination was performed as a result of industry experience with cracking caused by thermal cycling in similar valves. Sample plant personnel determined the affected valve remained operable based on an analysis that predicted very slow growth of this flaw.

The ISI Program is frequently updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the ISI Programs to reference more recent editions of ASME Section XI at the end of each inspection interval, ensures the ISI Program reflects enhancements due to operating experience that has been incorporated into ASME Section XI.

A sample plant NRC inspection reviewed the Inservice Inspection Program. No violations were identified and the implementation of the Program was found to meet ASME Code Requirements.

A review of NRC Inspection Reports, QA Audit/Surveillance Reports, and Self-Assessments since 1999 revealed no issues or findings that could impact the effectiveness of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program. As additional operating experience is obtained, lessons learned will be used to adjust this program, as needed.

Conclusion

The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program has been effective in managing aging effects, including loss of material due to corrosion, erosion or wear; cracking; and loss of mechanical closure integrity at bolted or welded connections due to wastage from borated coolant leakage, wear, or stress relaxation.

The continued implementation of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program provides reasonable assurance that the aging effects will be managed, such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B2.1.2 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

Program Description

The Periodic Surveillance and Preventive Maintenance Program is an existing plant-specific program that consists of the appropriate ten elements described in Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." The Periodic Surveillance and Preventive Maintenance Program manages aging effects for SSCs within the scope of license renewal. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation on a specified frequency based on operating experience. Leak inspections of piping and components in selected portions of systems are also performed on a specified frequency. Additionally, the program provides for replacement or refurbishment of certain components on a specified frequency, based on operating experience. The

Periodic Surveillance and Preventive Maintenance Program is also used to verify the effectiveness of other aging management programs.

Aging Management Program Elements

The key elements of aging management activities, which are used in the Periodic Surveillance and Preventive Maintenance Program, are described below. The results of an evaluation of each key element against the appropriate ten elements described in Appendix A of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," is provided below.

Scope of Program

The Periodic Surveillance and Preventive Maintenance Program manages aging effects for SSCs within the scope of license renewal. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation, on a specified frequency, based on operating experience. Leak inspections of piping and components in selected portions of systems are also performed on a specified frequency. Additionally, the program provides for replacement or refurbishment of certain components on a specified frequency, based on operating experience. The Periodic Surveillance and Preventive Maintenance Program is also used to verify the effectiveness of other aging management programs.

Preventive Actions

The Periodic Surveillance and Preventive Maintenance Program is a condition monitoring program. There are no preventive measures, as part of this program, associated with the aging effects of concern for license renewal. The visual inspection and examination of surfaces of selected equipment items and components, including fasteners, and leak inspections of piping and components in selected portions of systems on a specified frequency, are intended to identify the extent to which aging effects are occurring (i.e. condition). The replacement or refurbishment of certain components on a specified frequency does not prevent aging effects from occurring. These components are replaced or refurbished on a given frequency based on operating experience.

Parameters Monitored or Inspected

Surface conditions of selected equipment items and components, including fasteners, are monitored through visual inspection and examination for evidence of defects and age-related degradation, on a specified frequency, based on operating experience. Piping and components in selected portions of systems are monitored through visual inspection for evidence of leaks on a specified frequency. Certain components are replaced or refurbished on a given frequency based on operating experience.

Detection of Aging Effects

The aging effects of concern will be detected by visual inspection and examination of surfaces of selected equipment items, piping and components, including fasteners, for evidence of age-related degradation. Guidelines provided in the Westinghouse Aging Assessment Field Guides may be used as an aid in the identification of undesirable conditions.

The periodicity of most surveillance and preventive maintenance activities that are credited for license renewal will usually be driven by considerations other than aging, since the effects of aging usually occur slowly over time. For example, a check valve internal inspection is more likely to be driven by seat/disc/hinge pin wear than by erosion or corrosion of the valve body. Therefore, the specified frequencies of surveillance and preventive maintenance activities credited for license renewal may be adjusted or the performance of these activities deferred subject to the following constraints.

The frequency of surveillance and preventive maintenance activities that are credited for license renewal may be adjusted provided an engineering evaluation is performed justifying the revised frequency based on plant and industry operating experience.

Monitoring and Trending

The Periodic Surveillance and Preventive Maintenance Program is a condition monitoring program. Detailed material surface condition and leakage inspections and examinations, and component replacement or refurbishment activities are performed on a specified frequency based on operating experience. The results of these surveillance and preventive maintenance activities are documented, and subject to review and approval.

Acceptance Criteria

Acceptance criteria for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, and leak inspections of piping and components in selected portions of systems, are provided in the surveillance and preventive maintenance activities credited for license renewal. The acceptance criteria are related to the aging effect(s) of concern and are tailored to each individual inspection and examination, considering the aging effect(s) being managed.

Corrective Actions

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the guidance of ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as committed in the FSAR. Provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause determinations and prevention of recurrence where appropriate, are included in the corrective action program.

Corrective actions are implemented through the initiation of an Action Request in accordance with the corrective action program. Equipment deficiencies are corrected through the initiation of a Work Order in accordance with Plant procedures.

Confirmation Process

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the guidance of ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as committed in the FSAR. The aging management activities required by this program would also uncover any unsatisfactory condition due to ineffective corrective action.

The Action Request Process includes provisions for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken. The Action Request Process is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of an Action Request. The Periodic Surveillance and Preventive Maintenance Program includes provisions for verifying the completion and effectiveness of corrective actions for equipment deficiencies. This procedure establishes criteria for the selection and documentation of Post-Maintenance Tests (PMTs), guidelines to ensure equipment will perform its intended function prior to return to service, and guidelines to ensure the original equipment deficiency is corrected and that a new deficiency has not been created.

Administrative Controls

The Periodic Surveillance and Preventive Maintenance Program is implemented through various plant administrative procedures. These implementing documents are subject to administrative controls, including a formal review and approval process, in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," andthe guidance of ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as committed to in the FSAR.

Various procedures provide the required administrative controls, including a formal review and approval process, for procedures and other forms of administrative control documents.

Operating Experience

The Periodic Surveillance and Preventive Maintenance Program has been effective in maintaining the intended functions of long-lived passive SSCs, with an improving trend noted in the internal and external assessments performed over the past several years. Many Condition Reports, Action Requests and Work Orders have been generated and resolved through the implementation of this program, which demonstrates the effectiveness of this program to identify and correct age-related degradation prior to a loss of intended function. The effectiveness of this program is also demonstrated by the level of system/equipment availability as documented via the Maintenance Rule Periodic Assessments.

Enhancements

Program Elements Affected

Revise applicable existing procedures to ensure that the procedures address the following elements:

- **Detection of Aging Effects**
  Surveillance and preventive maintenance activities credited for license renewal aging management will be specified by call-ups maintained in the equipment database, flagged as license renewal commitments, and subject to additional requirements and controls, including the constraints placed on deferrals, cancellations and frequency changes for license renewal.

The frequency of surveillance and preventative maintenance activities credited for license renewal may be adjusted or the activity cancelled provided an engineering evaluation is performed justifying the revised frequency based on plant and industry operating experience.

- **Acceptance Criteria**
  Acceptance criteria shall be specified in the surveillance and preventive maintenance activities credited for license renewal. The acceptance criteria shall be related to the aging effect(s) of concern and tailored to each individual inspection and examination considering the aging effect(s) being managed.
- **Corrective Actions**
  Documents that implement aging management activities for license renewal shall direct that an Action Request (AR) be prepared in accordance with plant procedures whenever the acceptance criteria are not met.
- **Confirmation Process**
  The Periodic Surveillance and Preventive Maintenance Program shall be periodically audited by the Nuclear Oversight Group to insure its effectiveness and continued improvement.

Enhancements are scheduled for completion prior to the period of extended operation.

Conclusion

The Periodic Surveillance and Preventive Maintenance Program is an existing program. It uses, as its bases, various INPO and industry standards, including ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants."

The Periodic Surveillance and Preventive Maintenance Program has been effective in maintaining the intended functions of long-lived passive SSCs, with an improving trend noted in the internal and external assessments performed over the past several years. NRC Inspection Reports, QA Audit/Surveillance Reports, and Self-Assessments since 1999, and INPO evaluation reports were reviewed to assess the effectiveness of the Periodic Surveillance and Preventive Maintenance Program. Many surveillance and preventive maintenance activities were noted as being effectively performed.

Therefore, there is reasonable assurance that aging effects will be managed by the Periodic Surveillance and Preventive Maintenance Program such that SSCs within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B3.0 TLAA EVALUATION OF AGING MANAGEMENT PROGRAMS UNDER 10 CFR 54.21(C)(1)(iii)

B3.1 ENVIRONMENTAL QUALIFICATION PROGRAM

Program Description

The Environmental Qualification (EQ) Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered TLAAs for license renewal. The EQ Program ensures that these EQ components are maintained within the bounds of their qualification bases.

NUREG-1801 Consistency

The Environmental Qualification Program is an existing program, that was established to meet Plant commitments for 10 CFR 50.49. It is consistent with NUREG-1801, Section X.E1, "Environmental Qualification (EQ) of Electric Components" (Reference 3).

Exceptions to NUREG-1801

None.

Enhancements

None.

Operating Experience

The Environmental Qualification Program includes consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended function(s) during accident conditions after experiencing the effects of inservice aging. Based upon a review of the existing program and operating experience, the continued implementation of the Environmental Qualification Program provides reasonable assurance that the aging effects will be managed and that EQ components will continue to perform their intended function(s) for the period of extended operation.

Conclusion

Based upon a review of the existing program and operating experience, the continued implementation of the Environmental Qualification Program provides reasonable assurance that the aging effects will be managed and that the in-scope EQ components will continue to perform their intended function(s) for the period of extended operation. The

Environmental Qualification Program will also continue to be subject to periodic internal and external assessments to insure its effectiveness and continuous improvement. This result meets the requirements of 10 CFR 54.21(c)(1)(iii).

B4.0 REFERENCES

1. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 2001.

2. 10 CFR 50 Appendix B

3. NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* U.S. Nuclear Regulatory Commission, July 2001.

4. NRC SER, Sample Plant - Relief Requests RR 1-24 and RR-2-30 RE: Use Of ASME Code, Section XI, 1998 Edition With Addenda Through 2000, November 5, 2001.

Recommendations To Applicants

For

Enhancing NRC License Renewal Application (LRA) Review Efficiency

May, 2003

Introduction

Throughout the fall of 2002, NEI coordinated a nuclear industry effort where the NRC met periodically with representatives for those applicants who were planning to submit License Renewal Applications (LRAs) in 2003 and early 2004. The objective of the effort was to develop a Standard License Renewal Application (SLRA) format for Section 2, Section 3 and Appendix B. An example application, in the format for those sections, was sent to the NRC for review in January of 2003. The NRC commented on the SLRA, the comments were appropriately incorporated and a final version (May, 2003) was formally submitted to the NRC in August of 2003.

During the development of the Standard License Renewal Application (SLRA), a list was assembled documenting recommendations from NRC staff members for making future LRA reviews more efficient. These recommendations are contained within this document. Where appropriate, the recommendations were incorporated into the SLRA. Each incorporated recommendation is identified with its associated SLRA section number, inside of brackets, next to the item within the list.

The items in this document are only recommendations. They are not requirements. An applicant should evaluate the list and incorporate the recommendations that the applicant feels would best serve them during the NRC review of their application.

This document is divided into the following four sections:

- **General** = Items that pertain to the entire application or application process
- **Section 2.0** = Items that pertain only to Section 2.0
- **Section 3.0** = Items that pertain only to Section 3.0
- **Appendix B** = Items that pertain only to Appendix B

General

1. Dialog early with the staff on Fire Protection scoping. This is a complex area and may require additional time/clarification to ensure good communication between the reviewer and the applicant.

2. If NUREG-1801 or the SRP specifically identifies information that the applicant should provide, the LRA should have the information.

3. If possible, make a linkage between LRA sections and the FSAR on the LRA Compact Disk (C/D); and make sure the linkage works in all cases. This will significantly improve the efficiency of the referencing process.

4. If possible, provide the LRA table of contents ~60 days prior to LRA submittal. This would help the staff align its resources and prepare for the LRA submittal.

5. If possible, provide the staff with a list of the Aging Management Programs (AMPs) credited for License Renewal, along with a distribution table that indicates where each program is used (e.g., RCS, ESF, Auxiliary Systems, etc.) ~60 days prior to LRA submittal. This would help the staff align its resources and prepare for the LRA submittal.

Section 2.0

1. Address the License Renewal ISGs, and the applicant's approach to each, in the LRA. [Section 2.1.X – Interim Staff Guidance Discussion]

2. The system description section of the LRA should contain sufficient detail for the staff to use it in the section of the Safety Evaluation Report (SER) that discusses the technical information in the LRA. [Section 2.3.2.1 – System Description]

3. In the Chapter 2 System Descriptions, the staff would like to see a discussion of the system intended function (i.e., why the system is in scope for License Renewal). This is especially important for Auxiliary systems (and structures/structural groupings) where it isn't always clear from the FSAR description why the system (and structure/structural grouping) is within the scope of License Renewal. [Section 2.3.2.1 – System Description]

4. Make sure the system descriptions have clarity with respect to system boundaries. If the reason for the system boundary identified on the LRA print is not readily apparent, then the LRA should describe the reason for the system boundary (e.g., the portions of this system containing components subject to an AMR, extend from _____ to _____). Make sure boundary descriptions link back to system intended functions. [Section 2.3.2.1 – System Description]

5. System "realignment" occurs when:

    a. A system is defined by the applicant as out of scope for License Renewal, but components get put back in scope due to 10 CFR 54.4 (a)(2).

    b. Components have a system designation of one system, but since they have no License Renewal intended function in that system, the system is listed as out of scope; and the components of interest are listed as in-scope for the system they support.

    If system boundaries are realigned, the reason for the realignment, and an explanation of how it was accomplished, needs to be clearly stated in the LRA.

6. Identify all FSAR references in Section 2, not just one reference location. For several cases in the past, only one FSAR reference has been given in LRA Section 2, when the information actually exists in several sections of the FSAR; and staff reviewers have had to hunt down the additional information. This can be quite time consuming.

7. Do a consistency check between systems listed in the FSAR and those listed in the LRA. The staff has been finding some systems in the FSAR that weren't listed in the LRA.

8. Add a short sentence that indicates that the AMR results are shown in the corresponding Section 3 table and add a hyperlink, if possible. [Section 2.3.2.1 – Components Subject to AMR]

9. Add a table that describes the component intended functions. [Table 2.1-1]

Section 3.0

1. Section 3.0 should contain a description of the two table types and how they work together. [Section 3.0 – Table Description]

2. Make it clear that Table 3.x-1 is essentially the NUREG-1801, Volume 1, table with the addition of an "Item Number" column and a "Discussion" column. Note that the "Item Number" column allows the reviewer to align the Table 3.x-1 row with the corresponding NUREG-1801, Volume 1 table row to check consistency; and allows cross-referencing from Table 3.x.2-y. [Section 3.0 – Table 1 description]

3. Change the Section 3.2.1 title from "Scope" to "Introduction." [Section 3.2.1]

4. Make sure that corresponding columns in the Section 3 tables use the same terminology/labels. [Table 3.2.1 and Table 3.2.2-1]

5. In Section 3 of the application, add a summary description of the materials, environments, aging effects requiring management and aging management programs used for each system, structure group and electrical commodity group, that can be used in writing the Safety Evaluation Report (SER). Since the reviewer needs this information to develop the SER, it would most likely be more cost effective for the applicant to provide this description, than it would be for the reviewer to develop it. Additionally, the applicant may be able to develop a program that "automatically" assembles this information for inclusion in the LRA, making the effort even more cost effective. [Section 3.2.2.1]

Table 3.x-1

1. Add an LRA subsection reference to the applicable paragraphs in the "Discussion" column for those items where further evaluation is recommended. [Table 3.2.1]

2. Add all NUREG-1801, Volume 1 Row numbers for accounting. For those rows that are not applicable because of plant design, just add "BWR(PWR) Only" [Table 3.2.1]

3. If a plant specific program is designated in table 3.x-1, identify the program and its detailed description and justification location in the LRA (and add a hyperlink if possible). [Table 3.2.1]

4. When NUREG-1801, Volume 1 lists "Plant Specific" for a program, if all other items in that particular row in Table 3.x-1 are consistent with NUREG-1801, the entire row can be considered to be consistent with NUREG-1801. However, in the Table 3.x-1 "Discussion" column, the applicant must identify the program or programs being used to satisfy that NUREG-1801 line item. [Table 3.2.1]

5. If the words "Not Applicable" are used in the "Discussion" column, make it <u>very</u> clear what "Not Applicable" means. [Table 3.2.1]

Table 3.x.2-y

1. The use of Intended Functions abbreviations in Table 3.x.2-y is permissible (and even preferable) as long as the abbreviations are defined in the Intended Functions tables of Section 2. [Table 3.2.2-1 and Table 2.0-1]

2. For the standard LRA format, the Class of '03 should develop several examples of standard notes to be used that amplify whether a table row is consistent with NUREG-1801 or consistent w/exceptions, or plant-specific. The industry standard notes should be designated by a letter and the plant specific notes should be designated by a number; since it is anticipated that there will be many more plant specific notes than industry standard notes. [Table 3.2.2-1 – "Notes for Tables 3.2.2-1 through 3.2.2-X" immediately following Table 3.2.2-X]

3. The title of column #6, "GALL Item," should be changed to "NUREG-1801, Volume 2 Item" to be consistent with the Class of '03 recommendation to refer to GALL, within the application, as NUREG-1801. [Table 3.2.2-1]

Appendix B

1. Be cautioned that NUREG-1801 program X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," is intended to address environmentally-assisted fatigue, not other aspects of fatigue monitoring. Also, be advised that NUREG-1801 program XI.M19, "Steam Generator Tube Integrity," addresses steam generator tubes only, not other SG components.

2. NUREG-1801 AMP XI.M27, "Fire Water System," states that the AMP applies to underground piping, among other components. However, nowhere in the remainder of the NUREG-1801 description does it discuss how to manage the effects of aging on underground piping. This needs to be addressed.

3. If exceptions are taken to generic communications that are referenced in the NUREG-1801 AMPs, note this in the LRA and provide justification for why the plant AMP is still consistent with the NUREG-1801 AMP.

4. If credit is being taken for a Topical Report, make sure the topical report is applicable for 60 years.

5. Applicant Action Items of referenced WCAPs need to be addressed in the LRA.

6. Consider identifying, in a list, which programs are currently existing and which programs are new. [Section B1.5]

7. Each exception to NUREG-1801 should be clearly defined, explained, and justified. [Section B2.1.1]

8. State why exceptions to the NUREG-1801 program are necessary for the plant and include in which program element(s) the exceptions are reflected. Be sure to identify the exceptions within the actual program element descriptions. [Section B2.1.1]

9. State why enhancements to the NUREG-1801 program are necessary for the plant and include in which program element(s) the enhancements are reflected. Be sure to identify the enhancements within the actual program element descriptions. [Section B2.1.1]

10. Modify the title of Section B3.0 to say, "TLAA Evaluation of Aging Management Programs Under 10 CFR 54.2(c)(1)(iii)." "TLAA Support Activities" is not an accurate characterization of this section. [Section B3.0]

APPENDIX E

Interim Staff Guidance Documents

Note
ISGs are available on the NRC License Renewal Guidance Documents web page

APPENDIX F

Industry Guidance on Revised 54.4(a)(2) Scoping Criterion (Non-Safety Affecting Safety)

Industry Guidance on Revised 54.4(a)(2) Scoping Criterion (Non-Safety Affecting Safety)

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1 Purpose

The purpose of this document is to provide license renewal applicants with a consistent approach for addressing the non-safety affecting safety scoping criterion (10CFR54.4(a)(2)). Interpretations of this criterion have evolved since publication of the License Renewal Rule (10CFR54). The NRC has issued a generic request for additional information (RAI) on this topic, and followed it up with a letter stating the staff position. The industry discussion and guidance are based on positions taken by previous and near-term applicants, to resolve this issue.

2 NRC Staff Position on 54.4(a)(2) Scoping Criterion

The following is taken directly from the NRC letter (Ref. 7.1), in its entirety.

> *1. BACKGROUND*
>
> *Section 54.29 of 10 CFR Part 54 (the Rule) states that a renewed license may be issued by the Commission if the Commission finds that actions have been or will be taken with respect to the matters identified in 54.29(a)(1) and (a)(2) such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the CLB in order to comply with this paragraph are in accord with the Atomic Energy Act and the Commission's regulations. These matters include managing the effects of aging during the period of extended operation to assure the functionality of structures and components that have been identified to require review under Section 54.21(a)(1).*
>
> *The Statements of Consideration (SOC) for the Rule state that the objective of a license renewal review is to determine whether the detrimental effects of aging, which could adversely affect the functionality of systems, structures, and components (SSCs) that the Commission determines require review for the period of extended operation, are adequately managed.*
>
> *Section 54.4(a)(2) of the Rule states that all non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in Section 54.4(a)(1) should be included within the scope of the Rule. The SOC provides additional guidance related to this scoping criterion. Specifically, the SOC states that "To limit this possibility for the scoping category relating to non safety-related systems, structures, and components... An applicant for license renewal should rely on the plant's CLB, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those non safety-related systems, structures, and components that are the initial focus of the license*

renewal review. Consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required." (Federal Register, Volume 60, No. 88, 22467).

2. *DISCUSSION*

The SOC articulates the underlying philosophy of the Rule; that during the period of extended operation, safety-related functions should be maintained in the same manner and to the same extent as during the current license term.

The staff must have reasonable assurance that the applicant has identified all non safety-related SSCs that meet the 54.4(a)(2) scoping criterion. To accomplish this, the applicant should clearly describe the methodology used to determine those non safety-related SSCs that meet this criterion. This description should include how plant-specific failures of non safety-related SSCs and industry failures of such SSCs were considered in this determination, and should identify whether consideration was given to non safety-related SSCs which may not have failed during the current term, but may have a reasonable expectation of failure during the extended term. Such consideration should be based on sound engineering judgement that assures the failure of those non safety-related SSCs would not occur during the extended period of operation. Information which formed the basis for the applicant's conclusions need not be included in the application, but should be documented, auditable, and retrievable, in accordance with 10 CFR 54.37.

When demonstrating that failures of non safety-related SSCs would not adversely impact on the ability to maintain intended functions, a distinction must be made between non safety-related SSCs that are connected to safety-related SSCs and those that are not connected to safety-related SSCs. For a non safety-related SSC that is connected to a safety-related SSC, the non safety-related SSC should be included within the scope of license renewal up to the first seismic anchor past the safety/non-safety interface. Further, if the in-scope non safety-related structure or component is of the same commodity group (i.e., the same material/environment combination) as the safety-related structure or component to which it's connected, the same aging management programs should be applied to both the safety-related and non safety-related structures and components. If the in-scope non safety-related structure or component is not of the same commodity group, then aging management programs appropriate for the commodity should be applied.

For non safety-related SSCs which are not connected to safety-related piping or components or are beyond the first seismic anchor past the safety/non-safety interface, but have a spatial relationship such that their failure could *adversely impact on the performance of a safety-related SSC's intended function, the applicant has two options when performing its scoping evaluation; a mitigative option or a preventive option. With the mitigative option, the applicant should demonstrate that plant mitigative features (e.g., pipe whip restraints, jet impingement shields, spray and drip shields, seismic supports, flood barriers) are provided which protect safety-related SSCs from failures of non safety-related SSCs. This demonstration should show that the mitigating devices are adequate to protect safety-related SSCs from failures of non safety-related SSCs regardless of failure location (consideration can be given to the likelihood of failure at a particular location based on sound engineering judgement). If this level of protection can be demonstrated, then only the mitigative features need to be included within the scope of license renewal. However, if an applicant cannot demonstrate that the mitigative features are adequate to protect safety-related SSCs from the consequences of failures of non safety-related SSC's, then the applicant should utilize the preventive option, which requires that the entire non safety-related SSC be brought into the scope of license renewal. An applicant may determine that, in order to ensure adequate protection of the safety-related SSC, a combination of mitigative features and non safety-related SSCs must be brought within scope. Again, it is incumbent upon the applicant to provide adequate justification for the approach taken with respect to scoping of non safety-related SSCs in accordance with the Rule.*

To ensure that all relevant non safety-related SSCs are captured within the scope of the Rule, an applicant should consider not only its CLB, but also plant and industry operating experience. Operating experience includes all documented plant-specific and industry-wide experience that can be used to determine the plausibility of a failure. Documentation would include NRC generic communications and event reports, plant-specific condition reports, industry reports such as SOERs, and engineering evaluations.

3. CONCLUSION

On the basis of the guidance provided in the SOC, the staff expects applicants for license renewal to identify non safety-related SSCs whose failure could adversely impact intended functions. Such SSCs are to be included within the scope of license renewal. The evaluation to determine which non safety-related SSCs are within scope should not consider hypothetical failures, but should, based on engineering judgement and operating experience, consider the likelihood of system failure during the extended period of operation. The information used to support the scoping determination should be documented and available for staff review.

Based on the original Rule and the above guidance, components meeting the scoping criterion of 54.4(a)(2) will generally fall into three categories. 1) A plant's current licensing basis (CLB). The CLB will generally include a number of specific issues that meet the criterion of 54.4(a)(2). 2) Non-safety-related (NSR) SSCs directly connected to safety-related (SR) SSCs (typically piping systems). 3) NSR SSCs that are not directly connected to SR SSCs. In this case, two options are provided, a mitigative option or a preventive option.

The following discussion is intended to provide the rationale in these three categories, for determining which NSR SSCs would be considered within the scope of license renewal per 54.4(a)(2).

3 Non-Safety SSCs Typically Identified in the Current Licensing Basis

Non safety-related SSCs may have the potential to prevent satisfactory accomplishment of safety functions. Typical situations identified in the CLB where this can occur include but are not limited to the following.

3.1 Missiles

Missiles can be generated from internal or external events such as failure of rotating equipment. Inherent NSR features that protect safety-related equipment from missiles are within the scope of license renewal per 54.4(a)(2).

These protection features (missile barriers) are typically included as part of the building structure, and evaluated in the civil/structural area review.

3.2 Cranes

Most plants utilize a number of cranes in support of unit operations and maintenance activities that might be used to move heavy loads over safety-related equipment, spent fuel, or fuel in the core. Damaged spent fuel could release radioactive material potentially resulting in off site doses that exceed 10CFR100 limits. If the dropped heavy load damaged equipment associated with safe shutdown, the ability to achieve and maintain safe shutdown might be compromised. NUREG-0612 was issued by the NRC to provide guidelines to prevent heavy load drops that might affect safety-related equipment or cause fuel damage that would result in significant off site releases.

The overhead-handling systems, from which a load drop could result in damage to any system that could prevent the accomplishment of an SR function, are considered to meet the criteria of 54.4(a)(2) and are within the scope of license renewal.

3.3 Flooding

Flooding from various sources is generally considered during design of the plant. Typically, only equipment in the lowest levels of the plant is susceptible to flooding. (This assumes open stairwells and floor grating to allow floodwater to cascade to lower levels. If a room does not allow for cascading, it would need to be dispositioned on a plant-specific basis.) If level instrumentation and alarms are utilized to warn the operators of flood conditions, and operator action is necessary to mitigate the flood, then these instruments and alarms are within the scope of license renewal per 54.4(a)(2). If NSR sump pumps, piping and valves are necessary to mitigate the effects of a flood that threatens SR SSCs intended functions, then these components are also within the scope of license renewal per 54.4(a)(2).

Walls, curbs, dikes, doors, etc., that provide flood barriers to SR SSCs are within the scope of license renewal per 54.4(a)(2), and are typically included as part of the building structure, and evaluated in the civil/structural area review.

3.4 HELB

A high energy system is defined in each plant's CLB, either as a system that operates >200°F <u>and</u> >275 psig, or that operates >200°F <u>or</u> >275 psig. Typically, a plant will have evaluated all high energy systems outside containment in their High Energy Line Break (HELB) analysis. NSR whip restraints, jet impingement shields, blowout panels, etc., that are designed and installed to protect SR equipment from the effects of a HELB, are within the scope of license renewal per 54.4(a)(2). These protective features are typically associated with the structure and would be addressed in the civil/structural area review.

If the HELB analysis assumes that a NSR high energy piping system does not fail or assumes failure only at specific locations, then that piping system must be within the scope of license renewal per 54.4(a)(2), and subject to aging management review in order to ensure those assumptions remain valid.

4 <u>Non-Safety SSCs Directly Connected to Safety-Related SSCs</u>

For non-safety SSCs directly connected to safety-related SSCs (typically piping systems), the non-safety piping and supports, up to and including the first equivalent anchor beyond the safety/non-safety interface, are within the scope of license renewal per 54.4(a)(2). For this purpose the applicant must define the "first seismic or equivalent anchor" such that the failure in the non-safety related pipe run beyond the first seismic or equivalent anchor will not render the safety-related portion of the piping unable to perform its intended function under CLB design conditions. The applicant must be able to describe the structures and components that are part of the NSR piping segment up to and including the first seismic or equivalent anchor. The following apply.

4.1 A seismic anchor is defined as a device or structure that ensures that forces and moments are restrained in three (3) orthogonal directions.

4.2 An equivalent anchor may be defined in the CLB (i.e., UFSAR or other CLB documentation) and thus can be credited for the 10 CFR 54.4(a)(2) evaluation.

4.3 An equivalent anchor may also consist of a large piece of plant equipment (e.g., a heat exchanger) or a series of supports that have been evaluated as a part of a plant-specific piping design analysis to ensure that forces and moments are restrained in three orthogonal directions.

4.4 There may be isolated cases where an equivalent anchor point for a particular piping segment is not clearly described within the existing CLB information or original design basis. In those instances, the applicant may use a combination of restraints or supports such that the NSR piping and associated structures and components attached to SR piping is included in scope up to a boundary point that encompasses at least two (2) supports in each of three (3) orthogonal directions.

An alternative to specifically identifying a seismic anchor or series of equivalent anchors that support the SR/NS piping interface is to include enough of the NS piping run to ensure these anchors are included and thereby ensure the piping and anchor intended functions are maintained. The intended function consists of two facets 1) providing structural support for the SR/NS interface and 2) ensuring NS piping loads are not transferred through the SR/NS interface. The following methods (a) thru (d) are used to define end points for the portion of non-safety-related piping attached to safety-related piping to be included in the scope of license renewal. The bounding criteria in methods (a) thru (d) provide assurance that license renewal scoping for all plants encompass the NS piping systems included in the design basis seismic analysis and is consistent with the current licensing basis.

a. A base-mounted component (e.g., pump, heat exchanger, tank, etc.) that is a rugged component and is designed not to impose loads on connecting piping. The LR scope should include the base-mounted component as it has a support function for the SR piping.

Basis: A base-mounted component would either constitute an analysis endpoint as an anchor, or would be restricted from significant loading to or from the piping system based on the load carrying capacity of the component (such as a thin-walled tank). In the first case, the analysis endpoint and the LR boundary endpoint coincide such that the LR boundary envelops the analysis. For the second case, since the equipment is mounted to the structure and component loading is limited based on the component design, significant reactions cannot be transmitted to the piping system. The piping system support design would be required to provide adequate support prior to the component nozzle attachment. Therefore, the analysis endpoint is established prior to the piping system reaching the equipment nozzle, and the LR boundary (which includes the base-mounted component) envelops the analysis. When the LR boundary endpoint is established at a base-mounted component, the base-mounted component and supporting structure are included in the scope of license renewal.

b. A flexible connection is considered a pipe stress analysis model end point when the flexible connection effectively decouples the piping system (i.e. does not support loads or transfer loads across it to connecting piping).

Basis: Expansion joints and flexible hoses are designed such that significant piping system loads are not transferred across the connection. For this reason, the piping system is adequately supported to allow analysis endpoints to be established prior to the flexible connection. Therefore, establishing the LR boundary endpoint at flexible connections ensures that the analysis endpoint is enveloped.

c. A free end of NS piping, such as a drain pipe that ends at an open floor drain.

Basis: The piping analysis cannot continue past a free end of the piping run. Therefore, establishing the LR boundary endpoint at the free end of the NS piping run ensures that the analysis endpoint is enveloped.

d. For NS piping runs that are connected at both ends to SR piping include the entire run of NS piping.

Basis: All NS piping between the ends of the SR piping is conservatively included in the scope of license renewal ensuring the analysis endpoint is enveloped.

On a plant-specific basis, an applicant can elect to use methods (e) and (f) to define conservative end points for the portion of non-safety-related piping attached to safety-related piping to be included in the scope of license renewal. The plant-specific basis should be documented in a retrievable and auditable format and summarized in the license renewal application.

e. A point where buried piping exits the ground. The buried portion of the piping should be included in the scope of LR. A determination that the buried piping is well founded on compacted soil that is not susceptible to liquefaction must be made on a plant-specific basis.

Basis: The ground acts like an anchor if the buried piping is well founded on compacted soil that is not susceptible to liquefaction. Buried portions of piping runs for this condition are considered anchor points in the piping analyses. Since the analysis would consider the buried portion of piping as an anchor point, the establishment of the LR boundary endpoint where the piping run returns to above-grade ensures that the analysis endpoint is enveloped.

f. A smaller branch line where the moment of inertia ratio of the larger piping to the smaller piping is equal to or greater than the acceptable ratio defined by the current licensing basis, because significantly smaller piping does not impose loads on larger piping and does not support larger piping. The moment of inertia ratio must be determined on a plant-specific basis.

   Basis: The smaller diameter piping load carrying capacity is significantly less than that of the larger piping such that the larger piping is not adversely affected by the smaller line loads. The moment of inertia factor should be consistent with the plant-specific piping design basis. Therefore, establishing the LR boundary endpoint using this criterion ensures that the analysis endpoint is enveloped.

Extending the LR scope along the NS piping to these boundaries will ensure that NS piping up to a seismic anchor or equivalent anchor or the entire run of NS piping is included within the scope of LR.

5 Non-Safety SSCs Not Directly Connected to Safety-Related SSCs

For non-safety SSCs that are not directly connected to safety-related SSCs, or are connected downstream of the first equivalent anchor, the NSR SSCs may be in scope if their failure could prevent the performance of the system safety function for which the SR SSC is required. To determine which NSR SSCs may be in scope for 54.4(a)(2), two options exist: either a mitigative option or a preventive option.

5.1.1 Mitigative Option

An NRC reviewer described the mitigative option in a recent RAI (Ref. 7.4):

With respect to the mitigative approach, the applicant must demonstrate that plant mitigative features (e.g., pipe whip restraints, jet impingement shields, spray and drip shields, seismic supports, flood barriers, etc.) are provided which protect SR SSCs from a failure of NSR piping segments. When evaluating the failure modes of NSR piping segments and the associated consequences, age-related degradation must be considered. The staff notes that pipe failure evaluations typically do not consider age-related degradation when determining pipe failure locations. Rather, pipe failure locations are normally postulated based on high stress. Industry operating experience has shown that age-related pipe failures can, and do, occur at locations other than the high-stress locations postulated in most pipe failure analyses. Therefore, to utilize the mitigative option, an applicant should demonstrate that the mitigating devices are adequate to protect SR SSCs from failures of NSR piping segments at any location where age-related degradation is plausible. If this level of protection can be demonstrated, then only the mitigative *features need to be included within the scope of license renewal, and the piping segments need not be included within the scope.*

If an applicant [cannot demonstrate protection of the] *SR SSCs from the consequences of NSR pipe failures, then the applicant should utilize the preventive option, which requires that the entire NSR piping system be brought into the scope of license renewal and an AMR be performed on the components within the piping system.*

Finally, an applicant may determine that in order to ensure adequate protection of the SR SSC, a combination of mitigative features and NSR SSCs must be brought within scope. Regardless, it is incumbent upon the applicant to provide adequate justification for the approach taken with respect to scoping of NSR SSCs in accordance with the Rule. Therefore the applicant is requested to identify which option is used for NSR piping systems which are not connected to SR piping, but have a spatial relationship such that their failure could adversely impact on the performance of an intended safety function.
For each non-safety-related piping system which would normally be included within the scope of license renewal, but is excluded because mitigative features have been credited for protecting SR SSCs from the failure of the NSR piping system, please identify the following:

a. *the mitigative feature(s) that is credited for protection*
b. *the hazard (e.g., failure mechanisms and postulated failure locations) for which the mitigative feature(s) is providing protection*
c. *a summary discussion (including references, such as reports, analyses, calculations, etc.) of the basis for the conclusion that the mitigative feature(s) is adequate to protect SR SSCs.*

In this context, "mitigative" means that the effects of failures of an NSR SSC are mitigated by other SSCs. This mitigation is such that the failure of the NSR SSC will not prevent the performance of an SR SSC's intended function identified in 54.4(a)(1). If the mitigative option is used, then the mitigative features (whip restraints, spray shields, supports, barriers, etc.) need to be included within the scope of license renewal per 54.4(a)(2), and the non-safety system can be excluded from the scope of license renewal. These mitigative features are typically associated with the structure, and would be addressed in the Civil/Structural area review.

5.2 Preventive Option

If mitigative features are not installed, or cannot be shown to adequately protect safety related SSCs, then the preventive option needs to be used. The concern is that age-related degradation of non-safety SSCs could lead to interactions with safety-related SSCs that have not been previously considered. These interactions (pipe whip, physical impacts due to high energy system pipe falling due to FAC failures, jet impingement, spray or flooding from the non-safety systems) could create additional failures of the safety-related SSCs. The preventive option is where the most guidance is needed.

5.2.1 General Considerations

5.2.1.1 Loss of a Safety-Related Component vs. Loss of a Safety-Related Function

10CFR54.4 (a)(2) states that "*All non safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1) (i), (ii), or (iii) of this section*" are in scope. However, the NRC's Statements of Consideration have interpreted this to mean "*...(2) all non-safety related systems, structures, and components that support the function of a safety-related system, structure, or component or whose failure could prevent a safety-related system, structure, or component from satisfactorily fulfilling its intended function(s).*"

It is not considered acceptable to allow an NSR SSC to fail an SR SSC, on the basis that the SR intended function is maintained by redundant equipment. The only potential exception is if that position can be supported by the applicant's CLB.

An applicant may choose to defend (based on its CLB) that a loss of an SR SSC would not cause the loss of the SR system function. If an applicant chooses to utilize this position, it should be documented in a retrievable and auditable form.

5.2.1.2 Equipment Used to Establish Initial Conditions

For many plants, non-safety-related equipment, augmented with a suitable surveillance or monitoring program, is used to maintain safety-related equipment or plant conditions within limits consistent with event assumptions. As noted in the SOC for the license renewal rule, the Commission concluded that current activities for such systems, structures, and components, including licensee programs and the NRC regulatory process, are sufficient and that no additional evaluation is necessary for license renewal. NSR SSCs that are subject to plant Technical Specification Limiting Conditions for Operation are not within the scope of license renewal unless they meet the criteria of 54.4.

For instance, plant chemistry is assumed to be within the specifications maintained by the chemistry program based upon regular monitoring and analysis. Here, it is the monitoring or surveillance program that is primarily credited with ensuring the appropriate initial conditions exist, and the non-safety-related chemistry monitoring equipment is not in scope.

An additional point of reference is in Table 2.3-1 of NUREG-1800, where the failure of NSR cavity cooling HVAC ductwork "will not prevent the satisfactory completion of any critical safety function during and following a design basis event. Thus this ductwork is not within the scope of license renewal.

Therefore, the function of non-safety-related equipment to establish initial conditions for equipment operation or accident assumptions may not constitute the bases for inclusion in license renewal scope under 54.4(a)(2).

5.2.1.3 Malfunctions Resulting in Challenges to Safety-related SSCs

Malfunctions of non-safety-related equipment that result in a challenge to safety-related equipment do not constitute a basis for inclusion under §54.4(a)(2), since these malfunctions do not result in the loss of a safety-related function. For example, loss of a condensate pump might result in a reactor trip and resultant challenge to plant safety systems. However, this would not prevent the accomplishment of a function identified in 54.4(a)(1).

5.2.1.4 Cascading/Hypothetical Failures

The cascading issue applies to 10 CFR 54.4(a)(2) components and involves the consideration of subsequent levels of support systems that are necessary to ensure that a safety-related SSC performs it's intended function. For instance, the RHR pump seal coolers are cooled by the Service Water System, which performs as a second level support function in this capacity. The plant electrical system provides a third level support function in providing power to the Service Water pumps. The NRC staff's position on this issue is as follows:

"Therefore, to satisfy the scoping criterion under 10CFR 54.4(a)(2), an applicant needs to identify those non-safety-related SSCs (including certain second-, third- or fourth-level support systems) whose failure can prevent the satisfactory accomplishment of the safety-related function identified under 10CFR54.4(a)(1). In order to identify such systems, an applicant would consider those failures identified in 1) the documentation that makes up its CLB, 2) plant-specific operating experience, and 3) industry-wide operating experience that is specifically applicable to the facility. The applicant need not consider hypothetical failures that are not part of the CLB, and that have not been previously experienced." (Ref. 7.2)

Consistent with the staff's position, cascading must be considered to the same level that it is considered in the plant's CLB. Additionally, consideration will be given to plant specific and applicable industry operating experience to identify non-safety-related features that might be required to support the successful completion of a safety-related function.

5.2.2 System/Component Applicability

5.2.2.1 Systems and Components Containing Air/Gas

Air and gas systems (non-liquid) are not a hazard to other plant equipment. Industry operating experience (such as NUREG-1801, industry tools documents, and other LRA SERs) for systems containing air/gas, has shown no failures due to aging that have adversely impacted the accomplishment of a safety function. In addition, there are no credible aging mechanisms for air/gas systems with dry internal environments. A review of site-specific operating experience should be performed to verify this assumption. The results of this site-specific review should be maintained in a retrievable and auditable form. Additionally, components containing air/gas cannot adversely affect safety-related SSCs due to leakage or spray. Therefore, these systems are not considered to be in scope for 54.4(a)(2).

5.2.2.2 Systems Containing Liquids or Steam

5.2.2.2.1 High-Energy Systems

A high-energy system, without regard to seismic classification, is defined in each plant's CLB, either as a system that operates >200°F <u>and</u> >275 psig, or that operates >200°F <u>or</u> >275 psig. Physical impacts resulting from piping failures, pipe whip and jet impingement are credible only with high-energy systems. Industry experience has shown that physical impacts can occur due to high-energy piping failures caused by flow-accelerated corrosion. The effects of spray and harsh environment also need to be considered.

Non-safety high-energy piping with a potential for spatial interaction (pipe whip, jet impingement, physical impacts due to high energy system pipe failure due to FAC, spray or harsh environment) with vulnerable safety-related equipment that is not protected from the effects of a failure of the high energy line must be included within the scope of license renewal per 54.4(a)(2).

See Section 5.2.3 for definitions of vulnerable equipment.

5.2.2.2.2 Moderate/Low Energy Systems

Moderate/low energy systems, without regard to seismic classification, have potential spatial interactions of spray or leakage. Operating experience has shown that physical impacts from pipe whip, falling pipes or jet impingement from moderate-low energy systems do not occur and do not need to be considered. Industry experience indicates that piping does not fall if its supports are intact with the exception of failures of high-energy piping caused by flow-accelerated corrosion. Section 5.2.2.3 requires aging management of support systems that precludes physical impacts from moderate and low energy pipes falling.

Non-safety moderate/low energy piping that has potential spatial interactions (spray or leakage) with vulnerable safety related equipment that is not protected from the effects of spray or leakage must be included within the scope of license renewal per 54.4(a)(2).

See Section 5.2.3 for definitions of vulnerable equipment.

5.2.2.3 Non-Seismic and Seismic II/I Piping and Supports

This section is intended to describe the potential spatial interaction of non-safety piping systems that may fall on or otherwise physically impact safety related SSCs as a result of aging.

Reference 7.3 looked at earthquake experience data, including experience with aged pipe, and the following conclusions can be made:

- NO experience data exists of welded steel pipe segments falling due to a strong motion earthquake
- Falling of piping segment is extremely rare and only occurs when there is a failure or unzipping of the supports
- These observations hold for new and aged pipe

Piping supports for Seismic II/I piping need to be intact in order to prevent physical impacts on SR equipment during a seismic event and as a result must be included within the scope of license renewal per 54.4(a)(2).

Consistent with leak-before-break philosophy, it can also be assumed that piping that has retained its functional integrity will remain supported as long as its supports do not fail. If aged NSR piping has been shown to not fall during seismic events, it is logical to assume that it will also not fall as a result of only the aging process of the pipe except for FAC failures as demonstrated in NRC Information Notice 2001-09, as long as its supports are intact.

Therefore, as long as the effects of aging on the supports for these piping systems are managed, falling of piping sections, except for FAC failures, is not considered credible, and the piping section itself would NOT be in scope for 54.4(a)(2) due to the physical impact hazard (although the leakage or spray hazard may still apply).

All NSR supports for non-seismic or Seismic II/I piping systems with a potential for spatial interaction with safety-related SSCs, will be included within the scope of license renewal per 54.4(a)(2). These supports can typically be addressed in a commodity fashion, within the civil/structural area review.

Other potential physical impacts from swaying or other piping system movements due to seismic events or water hammer are not age-related, and therefore do not need to be considered in scope for 54.4(a)(2). Any of these potential physical impacts that are identified should be considered CLB/design issues and addressed via a plant's existing corrective action process. (Note: A plant-specific SQUG analysis may have addressed some of these issues.)

5.2.3 Vulnerability Clarifications

For an NSR SSC to be within the scope of license renewal per 54.4(a)(2), its failure due to age-related degradation must prevent the accomplishment of a SR SSC's intended function. An SR SSC is considered vulnerable if there are NSR SSCs in the vicinity whose failure could prevent accomplishment of the SR SSC's safety function, with the following clarifications.

5.2.3.1 Fail-Safe Components

Some safety-related components are fail-safe by design. Fail-safe components are components whose failure (through interaction with the failed NSR SSC) cannot prevent the accomplishment of the safety-related intended function. Fail-safe devices may not be vulnerable because their function may be accomplished as a result of their failure. As long as the NSR SSC failure causes the SR SSC to attain its fail-safe state, the NSR SSCs would NOT be considered in scope for 54.4(a)(2). If an applicant chooses to utilize this position, justification should be provided that failure of the NSR SSC would not result in a failure of the SR SSC to attain its fail-safe state. The current licensing basis, plant specific operating experience or industry operating experience can be used in the justification. If an applicant chooses to utilize this position, include the position in the LRA and document it in a retrievable and auditable form.

5.2.3.2 Components Qualified/Designed for Environment

If a component is qualified/designed to maintain its function in an environment that could be caused by failure of a nearby non-safety SSC, that non-safety SSC would NOT need to be within the scope of 54.4(a)(2). Assurance must be provided that the equipment's qualification/design is appropriate for all possible environments, before eliminating the non-safety system from scope.

6 Industry Guidance – Preventive Option

This section provides generic guidance for scoping under the preventive option. There are many different ways to achieve the desired result. When used, this guidance should be incorporated within plant specific rules and processes, and plant specific documentation should be developed.

A. Determine plant structures that house 54.4(a)(1) equipment.

B. Determine non-safety systems or portions of systems that are within the structures identified in A.

C. Determine vulnerable SR equipment (see Section 5.2.3) in the structures identified in A. If a plant participated in the SQUG effort, some of this information may already be available. However, the SQUG efforts typically only covered safe-shutdown paths and not all safety-related equipment/functions. Therefore, the plant specific SQUG evaluations need to be screened carefully.

D. Review documentation and/or perform walkdowns to identify non-safety systems or portions of systems that have spatial interaction potential with vulnerable equipment. Assume a failure anywhere along the length of the non-safety system. Use criteria developed in section 5.2.

E. Add these non-safety systems or portions of systems identified in D. to the scope of license renewal, and perform screening and aging management review, as appropriate.

F. Per 54.21(a)(2), describe and justify the plant specific methodology used to identify the 54.4.(a)(2) systems, structures and components requiring an aging management review following the guidance of NEI 95-10. The results from the application of this methodology should be plant specific (commodity lists, component lists, or boundary drawings, etc.), and included in the LRA and should be documented in a retrievable and auditable form including the bases for engineering judgments made during this review.

7 References

7.1 Letter from Grimes (NRC) to Nelson (NEI) and Lochbaum (UCS), regarding Guidance on the identification of Structures, Systems and Components that meet 10CFR54.4(a)(2), Dated March 15, 2002. This letter is a supplement to Reference 7.5.

7.2 Letter from Chris Grimes (NRC) to Doug Walters (NEI), Subject: "License Renewal Issue No. 98-0082, Scoping Guidance", dated August 5, 1999.

7.3 NUREG CR-6239 "Survey of Strong Motion Earthquake Effects on Thermal Power Plants in California with Emphasis on Piping Systems".

7.4 Letter from Robert Prato to David Christian, Request for Additional Information for the Review of the North Anna Nuclear Station, Units 1 and 2, and Surry Nuclear Station, Units 1 and 2, License Renewal Application, dated 10/22/01.

7.5 Letter from Grimes (NRC) to Nelson (NEI) and Lochbaum (UCS) regarding Scoping of Seismic II/I Piping Systems, dated Dec. 3, 2001.

7.6 Letter from Kuo (NRC) to Nelson (NEI) regarding Staff comments on Industry Guidance for 54.4(a)(2), dated March 21, 2003.